76. 038 EG2.4.11 SRO 001

Given the following plant conditions:

- The plant is operating at 100% RTP.
- The crew is implementing AOP-035, S/G Tube Leak, due to an identified tube leak in "B" S/G.
- Two charging pumps are running at maximum output and all letdown flowpaths have been isolated.
- RO reports that PZR Level is at 22% and continuing to lower in an uncontrolled manner.

Which ONE (1) of the following is the proper procedural progression for the given conditions?

EPP-7, SI Termination
EPP-12, Post-SGTR Cooldown Using Backfill

A.✓ Trip the Reactor --> Manual SI --> PATH-1 --> PATH-2 --> EPP-12

SI Termination

B. Trip the Reactor --> Manual SI --> PATH-1 --> PATH-2 --> EPP-7 --> EPP-12

C. Continue in AOP-035 --> Trip the Reactor and actuate SI when PZR level reaches 10% --> PATH-1 --> PATH-2 --> EPP-12

D. Continue in AOP-035, start the remaining charging pump and raise speed to maximum and monitor PZR level.

The correct answer is A.

A. Correct.

B. Incorrect. EPP-7 is not entered for a tube rupture. PATH-2 and EPP-12 will provide direction on when to secure SI pumps and resetting of ECCS components.

C. Incorrect. Foldout "A" does contain requirements to actuate SI if PZR Level cannot be maintained greater than 10%. AOP-035 does not provide this direction. AOP-035 requires that the reactor be tripped if RCS level is lowering uncontrollably with two charging pumps at maximum speed and letdown isolated.

D. Incorrect. This was the correct process up to a few years ago. In the past the third charging pump was started and taken to maximum speed after letdown was isolated. This was removed during a recent revision as a prudent action to trip the reactor with only two charging pumps running at maximum speed and letdown isolated.

Question 76
Tier/Group 1/1
K/A Importance Rating - RO 4.0 SRO 4.2

Steam Generator Tube Rupture: Knowledge of abnormal condition procedures.

Reference(s) - Sim/Plant design, AOP-035, PATH-1/2, EPP-12, EPP-7
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-035-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13

Comments - Discussed with P. Capehart on 6/15/11: Knowledge of abnormal condition procedures relative to a SGTR. At RNP a SGTR is mitigated utilizing the EOP network procedures. A S/G Tube Leak is mitigated using an AOP. Agreed that "abnormal condition procedures" also included the EOP network procedures

SRO: Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or which to proceed.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
6.	Adjust Charging Flow As Follows:	
	a. Check Charging Pump Status - AT LEAST TWO RUNNING	a. Start one additional Charging Pump.
	b. Place running Charging Pumps Speed Controllers in MAN AND adjust output to maximum	
7.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
8.	Check Letdown - IN SERVICE	Go To Step 11.
9.	Verify All Letdown Flowpaths Isolated As Follows:	
	• LCV-460A & B, LTDN LINE STOP Valves - CLOSED	
	• HIC-137, EXCESS LTDN FLOW Controller - ADJUSTED TO 0%	
	• CVC-387, EXCESS LTDN STOP - CLOSED	
10.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
11.	Trip The Reactor AND Go To Path-1 OR EOP-E-0, Reactor Trip or Safety Injection	
12.	Control Charging Flow To Maintain Desired RCS Level	
*13.	Check RCS Leakage - GREATER THAN RUNNING CHARGING FLOW	IF leakage exceeds Charging flow, THEN Go To Step 6. Go To Step 15.
14.	Go To Step 6	
15.	Notify Chemistry Personnel To Periodically Sample All S/Gs For Activity And Boron Concentration	

8.3.2 (Continued)

7. While the immediate actions are being performed, the CRS should verify their completion prior to entering the Path or applicable procedure. Once this verification is completed, the CRS should enter the procedure to verify the immediate actions as discussed above.

8. The Operators should broadcast the performance of immediate actions as they are being performed and then use 3-way communications to verify that immediate actions have been performed when the CRS uses the procedure.

9. Broadcasting of the Reactor Trip function should include the following as applicable:
   - Reactor Trip and Bypass Breakers Open
   - Rod Position Indication at zero
   - Rod Bottom Lights Illuminated
   - Neutron Flux decreasing

10. Broadcasting of the Safety Injection function should include the following as applicable:
    - Pressurizer Pressure is greater than 1715 psig.
    - No high steam line D/P or high steam flow bistables
    - CV pressure is less than 4 psig
    - Pressurizer level is greater than 10%, stable or rising
    - Safety Injection is not initiated and not required.

8.3.5 Precisely Controlling the Plant

1. A control band (upper and lower thresholds) is to be provided when a parameter is being controlled manually or is out of specification. The control band is established such that the plant is being controlled precisely. Supplemental monitoring and contingencies are to be established as required.

2. An actionable limit is to be set as a contingency.

3. The Shift Manager and Shift Technical Advisor are to provide oversight and ensure the CRS has established a control band with an actionable limit.

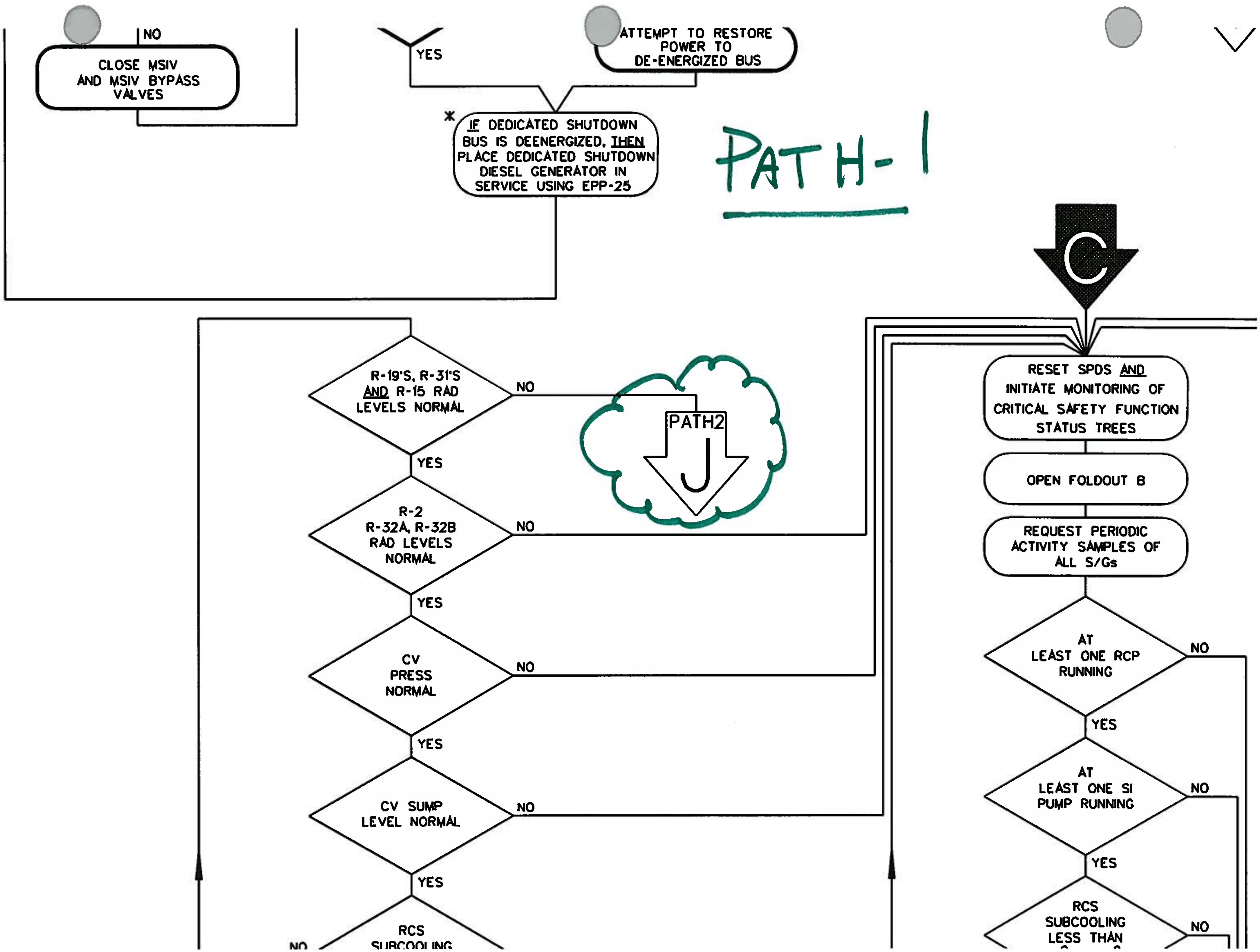
4. The CRS is to maintain Command and Control by establishing a control band with an actionable limit, supplemental monitoring, and contingencies.

5. The Reactor Operators are to operate equipment with established control bands with an actionable limit. The RO is to advise the CRS if the control band can **NOT** be maintained prior to exceeding the thresholds. Also, the RO is to provide input on what supplemental monitoring and contingencies are warranted to precisely control the plant.

8.3.6 Automatic Actions/Actuations (RAIL 94R0928)

1. During the course of an event, should the setpoint for an automatic protective system actuation be approached, the Operator should, if possible, manually initiate the actuation prior to the automatic actuation. If immediate actions are in progress they should be completed prior to initiating the signal, however this is not considered performance of steps early or out of order.

   Example: During an RCS leakage transient, after entry to the EOP Network, pressure is slowly decreasing and after observing the trend in RCS pressure it is apparent that RCS makeup can not keep up with leakage. As pressure approaches the low pressure SI setpoint of 1715 psig, the Operator should manually initiate Safety Injection prior to reaching the setpoint.



NO
CLOSE MSIV AND MSIV BYPASS VALVES
YES
ATTEMPT TO RESTORE POWER TO DE-ENERGIZED BUS
* IF DEDICATED SHUTDOWN BUS IS DEENERGIZED, THEN PLACE DEDICATED SHUTDOWN DIESEL GENERATOR IN SERVICE USING EPP-25
PATH-1
C
RESET SPDS AND INITIATE MONITORING OF CRITICAL SAFETY FUNCTION STATUS TREES
OPEN FOLDOUT B
REQUEST PERIODIC ACTIVITY SAMPLES OF ALL S/Gs
AT LEAST ONE RCP RUNNING
AT LEAST ONE SI PUMP RUNNING
RCS SUBCOOLING LESS THAN
R-19'S, R-31'S AND R-15 RAD LEVELS NORMAL
PATH2
J
R-2 R-32A, R-32B RAD LEVELS NORMAL
CV PRESS NORMAL
CV SUMP LEVEL NORMAL
RCS SUBCOOLING
YES
NO

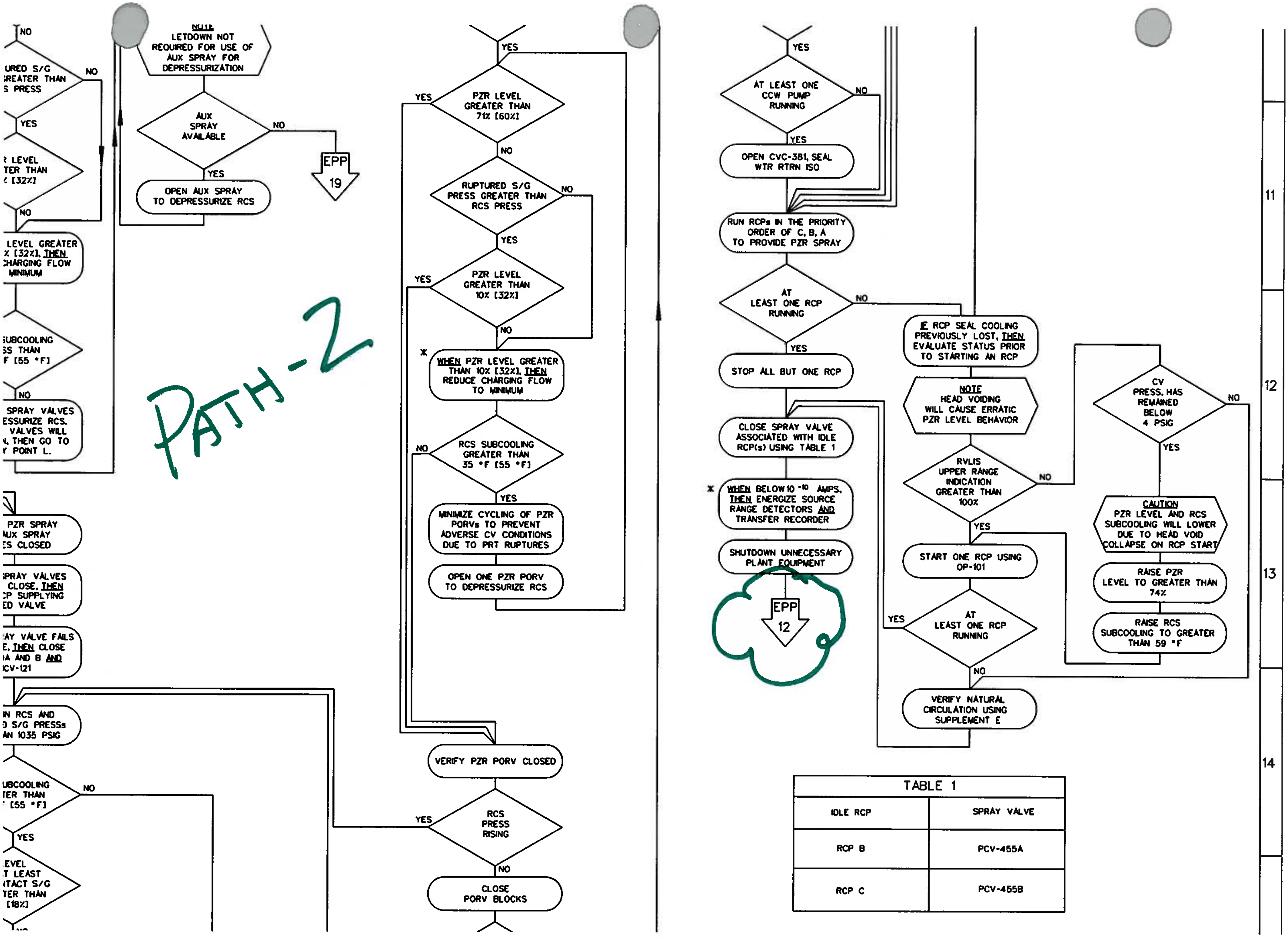


TABLE 1

IDLE RCP	SPRAY VALVE
RCP B	PCV-455A
RCP C	PCV-455B

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

END PATH PROCEDURE

EPP-12

POST-SGTR COOLDOWN USING BACKFILL

REVISION 14

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to cooldown and depressurize the plant to Cold Shutdown conditions following a Steam Generator Tube Rupture. This recovery method depressurizes the ruptured S/G by draining it through the ruptured tube into the RCS.

2. ENTRY CONDITIONS

   a. Path-2

   b. EPP-13, Post-SGTR Cooldown Using Blowdown, when S/G blowdown is not available and the backfill method of cooldown is selected.

- END -

77. 054 AG2.4.11 SRO 001
Given the following plant conditions:

- The plant is currently at 100% RTP with FWUFM in service.
- A feedwater flow transient has occurred.
- HCV-1459, LP Heaters Bypass Valve, is observed to be OPEN with its control switch in AUTO.
- Main Feedwater Pump suction pressure is currently 375 psig.

Which ONE (1) of the following completes the statements below?

The CRS will direct that Reactor Power be maintained less than 100% using __(1)__ or OMM-001-2, Shift Routine and Operating Practices.

IAW OMM-001-2, Reactor Thermal Power is allowed to increase to a maximum of __(2)__ MWth briefly during power excursions. RO knowledge

A.✓ (1) AOP-010, Main Feedwater/Condensate Malfunction, Attachment 1

(2) 2346 MWth

B. (1) OP-105, Maneuvering the Plant When Greater Than 25% Power

(2) 2385 MWth — why plausible? 102% of rated when used?

C. (1) AOP-010, Main Feedwater/Condensate Malfunction, Attachment 1

(2) 2385 MWth

D. (1) OP-105, Maneuvering the Plant When Greater Than 25% Power

(2) 2346 MWth

The correct answer is A.

A. Correct.

B. Incorrect. With HCV-1459 open the feedwater will become cooler due to bypassing the LP FW Heaters. This will cause reactor power to increase. The second part of the distractor is correct.

No

C. Incorrect. The first part of the distractor is correct. AOP-010 does direct the operators to control reactor power to less than 100%. Power will be lowered so that MFP suction pressure is greater than 400 psig. At 400 psig MFP suction pressure HCV-1459 should automatically open with the control switch in AUTO.

D. Incorrect. With HCV-1459 open the feedwater will become cooler due to bypassing the LP FW Heaters. This will cause reactor power to increase. AOP-010 does direct the operators to control reactor power to less than 100%. Power will be lowered so that MFP suction pressure is greater than 400 psig. At 400 psig MFP suction pressure HCV-1459 should automatically open with the control switch in AUTO.

Question 77
Tier/Group 1/1
K/A Importance Rating - RO 4.0 SRO 4.2

Loss of Main Feedwater (MFW): Knowledge of abnormal condition procedures.

Reference(s) - Sim/Plant design, AOP-010, OMM-001-2
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-010-006
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.13
Comments -

SRO: Assessing plant conditions and then selecting a section of a procedure to mitigate, recover, or with which to proceed. Also knowledge of maximum licensed thermal power limitations.

No

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Failure of HCV-1459, LP HEATERS BYP, will cause Reactor Power to rise. Steady State operation above 100% is NOT permitted.

****

NOTE

- HCV-1459, LP HEATERS BYP, is interlocked with QCV-10426, SECONDARY BYPASS. HCV-1459 failing open will also result in QCV-10426 opening.
- Rapid power reductions may result in the axial flux difference exceeding the operating band values and require a power reduction to less than 50% to comply with ITS 3.2.3 Condition C.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
35.	Determine If HCV-1459 Has Failed As Follows:	
	a. Check HCV-1459, LP HEATERS BYP - OPEN	a. Go To Step 38.
	b. Maintain Reactor Power less than 100% using Attachment 1	
	c. Continue to reduce power using Attachment 1 until Feed Water Pump suction pressures are greater than 400 psig. (Local Indication) • PI-I433 - "A" FW PUMP SUCTION PRESSURE • PI-I434 - "B" FW PUMP SUCTION PRESSURE	
	d. Check Feed Water Pump suction pressures - GREATER THAN 400 PSIG	d. WHEN Feed Water Pump suction pressures are greater than 400 psig, THEN Go To Step 35.e.
	e. Verify HCV-1459 - CLOSED	e. Locally isolate HCV-1459 by closing C-52A, HCV-1459 INLET.

9.3 Continuous Calorimetric Program

NOTE:	Power level is limited to a maximum of 100.3% (2346 MWth) per LDCR 02-0012, Appendix K Power Uprate.

5. During Steady State Operations, reactor power may indicate greater than the allowed RTP for brief periods of time with no operator action required. It is **NOT** intended that this flexibility be used to make up for lost generation when the Period Average falls significantly below Target Power level for extended periods. These periods should be limited as follows:

   a. The allowed thermal power with FWUFM in service is 2339MWth. With FWUFM out of service for longer than allowed by TRM 3.25, thermal power will be limited to 2300 MWth. The setpoints at which a power limit warning will be received are as follows: (RNP2 6004-CALO-SRS-001

      - The 8 hour average power is >2339 MWt and less than 1 hour remains in the 8 hour period
      - The 1 minute average power is >2339.35 MWt (100.015%) for 290 minutes continuously.
      - The 1 minute average power is >2340.75 MWt (100.075%) for 50 minutes continuously.
      - The 1 minute average power is >2342.50 MWt (100.150%) for 20 minutes continuously.
      - The 1 minute average power is >2346.00 MWt (100.299%) for 5 minutes continuously
      - The 4 minute average power is >2346.00 MWt (100.299%) instantaneously

   b. The 4 minute average thermal power indication should not be allowed to consistently exceed the allowed RTP without taking operator action to reduce power. Small fluctuations above and below the allowed thermal power limit are expected for short durations and are part of steady state operations. The magnitude, trend and average of the fluctuations must be evaluated to determine if a power reduction is required.

   c. The one hour average thermal power indication should not be allowed to exceed the allowed thermal power. It is recognized that the one hour average thermal power could exceed the allowed thermal power at the beginning of a new 8 hour period. It is not intended that a power reduction be initiated immediately in this condition. The trend should be evaluated and power reductions initiated if required.

6. During Planned Evolutions, operators should consider reducing power in advance of planned plant evolutions that have a potentially high likelihood of causing power to increase above the licensed power limit. **IF** during the evolution, power increases above the licensed power limit, **THEN** operators should take action to restore power to or below the licensed limit.

9.3 Continuous Calorimetric Program

7. Upon receiving any valid calorimetric Power Alarm, Reactor power shall be reduced to less than 2339 MWth, or until the calorimetric Power Alarm is cleared using the Valve Position Limiter. A time/power target will be given. Power must be reduced to less than the target power by the time given to prevent exceeding the limits. The cause of the Time / Power alarm may be determined by clicking on the ALARM INFORMATION button. **[R5] [R6]**

8. No information will be displayed for periods in which values cannot be calculated. For example, **IF** ERFIS is OOS for greater than thirty minutes, **THEN** no Period Average, Projected Power, or Recommended Power will be calculated. For periods in which ERFIS data is unavailable for less than thirty minutes **AND** the power change for the period ERFIS was OOS is less than 1%, the CCP will generate missing data using linear interpolation. **IF** it is anticipated that the Period Average will **NOT** be calculated, **THEN** the Reactor should be operated at less than 2339 MW Thermal. **WHEN** the CCP is OOS, **THEN** Reactor power shall be maintained less than or equal to 100% as indicated by other diverse indications to ensure Time / Power limits are not exceeded.

9. Other conditions may exist which effect the validity of the CCP calculation. **WHEN** these conditions exist, **THEN** other indications of Reactor power shall be used. Conditions which affect the validity of the CCP calculation include:
   - Operation at less than 15% power will cause some inputs to the CCP calculation to be outside reasonability limits.
   - **IF** Charging temperature or Letdown flow data is unavailable, **THEN** the CCP calculation will default to data for operation with only the 60 gpm orifice in service. Operation with greater than 60 gpm letdown flow will cause the calculation to be non-conservative.
   - Feeding any S/G with a MDAFW Pump will result in non-conservative results.

10. FWUFM measures feed flow and assumes all that flow (minus steam generator blowdown) is converted to steam flow, it instantly responds to changes in feed flow. Evolutions/events that perturb feed flow or temperature, such as manual FRV control, SDAFW Pump flow, placing/removing Condensate Polishers in/from service, removing S/G Blowdown from service, or sudden changes in ambient temperature particularly associated with heavy rain will cause FWUFM to indicate power has increased, when in fact steam demand (which reactor power follows) hasn't changed.

11. Temporary conditions that affect **OR** known to cause power as calculated by LEFM to increase above 2339 MWth should be monitored by diverse redundant indications with no action for up to 5 minutes to allow LEFM to stabilize. **IF** at any time a diverse redundant power indication shows an actual transient, **OR** changes in Steam Flow, Steam Pressure, **OR** unexplained secondary annunciators, **THEN** action should be taken to maintain reactor power ≤2339 MWth using the Valve Position Limiter. Steam Flow Calorimetric is an effective tool to validate LEFM response. LEFM increases that are solely a result of increases in feed flow would not be seen as a power increase using Steam Flow Calorimetric. **[R5] [R6]**

12. **IF** questions arise concerning the validity of the CCP calculation **OR** reactor power has exceeded the licensed power maximum of 2346 MWt, **THEN** contact Engineering **AND** initiate an NCR.**[R11]**

78. 055 EA2.01 SRO 001

Given the following plant conditions:

- The plant is in MODE 5 with RHR Pump "A" in service.
- The SUT has failed and both EDGs have tripped.

Which ONE (1) of the following completes the statement below?

FCV-605 and HCV-758 will fail __(2)__ and the conditions are required to be mitigated by implementing AOP-017, Loss of Instrument Air, Section C, RHR Aligned for Core Cooling, AND __(1)__.

A. (1) SHUT

(2) EPP-1, Loss of All AC Power

B. (1) OPEN

(2) EPP-1, Loss of All AC Power

C. (1) OPEN

(2) AOP-020, Loss of Residual Heat Removal (Shutdown Cooling)

D✓ (1) SHUT

(2) AOP-020, Loss of Residual Heat Removal (Shutdown Cooling)

E-Plan

K/A not tested at SRO

REVISE

The correct answer is D.

A: Incorrect - Both FCV-605 and HCV-758 fail SHUT on loss of IA. AOP-017 will be entered for a loss of instrument air, which will eventually occur during a station blackout. AOP-017 is a concurrent AOP, however, EPP-1 contains a note that explicitly states that Foldouts and concurrent AOPs should not be implemented during EPP-1. Also, IAW OMM-022, Emergency Operating Procedures User's Guide, Attachment 10.1, EOP Applicability Table, EPP-1 is not applicable when RHR is in service.

B: Incorrect - Both FCV-605 and HCV-758 fail SHUT on loss of IA. AOP-017 would be entered for a loss of instrument air, which will eventually occur during a station blackout. AOP-017 is a concurrent AOP, however, EPP-1 contains a note that explicitly states that Foldouts and concurrent AOPs should not be implemented during EPP-1. Also, IAW OMM-022, Emergency Operating Procedures User's Guide, Attachment 10.1, EOP Applicability Table, EPP-1 is not applicable when RHR is in service.

C: Incorrect - Both FCV-605 and HCV-758 fail SHUT on loss of IA. Second part of distractor is correct.

D: Correct - Both FCV-605 and HCV-758 fail SHUT on loss of IA. AOP-017 would be entered for a loss of instrument air, which will eventually occur during a station blackout. AOP-017 is a concurrent AOP, however, EPP-1 contains a note that explicitly states that Foldouts and concurrent AOPs should not be implemented during EPP-1. Also, IAW OMM-022, Emergency Operating Procedures User's Guide, Attachment 10.1, EOP Applicability Table, EPP-1 is not applicable when RHR is in service.

Question 78
Tier/Group 1/1
K/A Importance Rating - RO 3.4 SRO 3.7

Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system.

Reference(s) - Sim/Plant design, AOP-017, Pages 3 and 36; SD-003, RHR, Figure 3., OMM-022.
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-017-006
Question Source - BANK (Similar question used on 2008 NRC Exam. The question format and distractors have been significantly modified.)
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments -

SRO: The candidate must have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. OMM-022, Emergency Operating Procedures User's Guide, contains the instructions for the use of Emergency and Abnormal Operating Procedures.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

1. PURPOSE

This procedure provides the instructions necessary to mitigate the loss of RHR in all conditions for which RHR can be aligned to provide shutdown cooling. This includes loss of RHR cooling for reasons such as RCS leakage, loss of power, loss of Service Water or Component Cooling Water, RHR pump cavitation, and inadequate RHR flow or abnormal reductions in RHR cooling.

This procedure is applicable in Modes 4, 5, and 6 when fuel is in the vessel.

2. ENTRY CONDITIONS

Direct entry from any condition resulting in a loss of RHR pump(s), RHR pump cavitation, abnormal RHR flow or temperature control, loss of instrument bus, or excessive loss of RCS inventory while RHR is aligned for shutdown cooling.

As directed by the following other procedures:

- AOP-005, Radiation Monitoring System, when a low level in the SFP exists due to an RCS leak with the SFP GATE VALVE open.
- AOP-014, Component Cooling Water System Malfunction, resulting in stopping of the RHR Pumps while in CSD.
- AOP-016, Excessive Primary Plant Leakage, if less than 200°F and leakage exceeds Charging Capacity.
- AOP-017, Loss Of Instrument Air, if the loss of Instrument Air has affected core cooling while on RHR.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION C

RHR ALIGNED FOR CORE COOLING

(Page 1 of 6)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Determine If IA Capacity Has Been Restored As Follows:	
	a. Check IA Header pressure: • GREATER THAN 85 PSIG AND • STABLE OR RISING	a. IF IA capacity is restored, THEN Go To Step 1.b. Go To Step 2.
	b. Go To Attachment 4, Restoration From Loss Of Instrument Air	
2.	Check RHR Flow OR Temperature Control - ADVERSELY AFFECTED BY LOSS OF IA	IF RHR flow OR temperature control is affected, THEN perform Steps 3 and 4. Go To Step 5.
3.	Check RCS Temperature - LESS THAN 212°F	Control RCS temperature by dumping steam using one of the following methods listed in order of preference: a. Steam Dump to Condenser b. Steam Line PORVs controlled by IA c. Steam Line PORVs controlled by nitrogen using Attachment 2 d. Manually steam each intact S/G using Attachment 3
4.	Perform AOP-020, Loss Of Residual Heat Removal (Shutdown Cooling), While Continuing With This Procedure	
5.	Check RCS Status - SOLID PLANT	Go To Step 8.

ATTACHMENT 1

MAJOR COMPONENTS AFFECTED BY LOSS OF IA

(Page 3 of 5)

6. Isolation Valve Seal Water System Components FAIL POSITION

   a. PCV-1922 A & B, IVSW AUTO HEADER ISOLs - OPEN

7. Main Steam System Components FAIL POSITION

   a. MAIN STEAM ISOLATION VALVES - CLOSED

   b. STEAM LINE PORVs - CLOSED

8. Primary Sample System Components FAIL POSITION

   a. PS-956 A through H, PRIMARY SAMPLE ISOLATIONS - CLOSED

9. Radiation Monitoring System Components FAIL POSITION

   a. RMS-1,2,3 & 4, R-11/R-12 ISOL VALVES - CLOSED

10. Reactor Coolant System Components FAIL POSITION

   a. PCV-455 A & B, PZR SPRAYS - CLOSED

   b. RC-516 & 553, PRT TO GAS ANALYZER - CLOSED

   c. RC-519 A & B, PW TO CV ISOs - CLOSED

   d. RC-544, RV FLANGE LEAKOFF - OPEN

   e. RC-550, PRT NITROGEN SUPPLY - CLOSED

11. Residual Heat Removal System Components FAIL POSITION

   a. HCV-142, PURIFICATION FLOW - CLOSED

   b. HCV-758, RHR HX DISCH FLOW - CLOSED

   c. FCV-605, RHR HX BYPASS FLOW - CLOSED

ATTACHMENT 10.1
Page 1 of 3

EOP APPLICABILITY TABLE

PROCEDURE	WHEN APPLICABLE*	COMMENTS
PATH 1 PATH 2	> 350°F	Assumes RHR System not in service and SI operable. Modifications necessary if otherwise.
EPP-1 EPP-2 EPP-3 EPP-22	Modes including full power until RHR in service	Assumes RCS partly hot and pressurized. Problems with RCP seals are minimal if RCS is cold and depressurized.
EPP-25	All modes	EPP-25 can be used at any time desired to power equipment from the DSDG or to initiate backfeed.
EPP-4	Reactor Critical or at power	Assumes trip from power. Modifications necessary if otherwise.
EPP-5 EPP-6	>350°F	Assumes hot (near no load) conditions. Slight modification required if already in cooldown.
EPP-7	>350°F	Assumes RHR System not in service. Modifications necessary if otherwise.
EPP-8 EPP-9 EPP-10 EPP-Supplements	>350°F	Assumes RHR System not in service; entry is limited by stated conditions. Modifications necessary if otherwise.
EPP-11	>200°F	Assumes T_{hot} >212°F. S/Gs will not steam if below boiling point.
EPP-Foldouts	>200°F	Applicability dependent upon procedures that provide entry conditions.

* Condition at time of initiating event.

79. 058 AA2.03 SRO 001

The plant was operating at 100% RTP when the following indications are received:

- "B" RCP Red and Green lights extinguished.
- Reactor Trip Breaker "B" Red and Green lights extinguished.
- Channel #3 Reactor Protection Bistables illuminated.
- "B" MDAFW Pump Red and Green lights extinguished.
- Tave is being maintained at No-Load Value.

Which ONE (1) of the following identifies (1) the correct procedural progression and (2) the correct actions to be taken?

Enter PATH-1 --> __(1)__

Request a crew to locally trip OCBs 52/8 AND 52/9, __(2)__.

A.✓ (1) EPP-4, Reactor Trip Response --> EPP-27, Loss of DC Bus "B"

(2) Isolate Letdown, Reduce charging to minimum, control pressurizer pressure using control and backup heaters and PCV-456, PZR PORV

B. (1) EPP-4, Reactor Trip Response --> EPP-27, Loss of DC Bus "B"

(2) Verify FCV-626 is open and then secure all charging pumps, control pressurizer pressure using control and backup heaters and normal spray flow

C. (1) EPP-7, SI Termination --> EPP-27, Loss of DC Bus "B"

(2) Isolate Letdown, Reduce charging to minimum, control pressurizer pressure using control and backup heaters and PCV-456, PZR PORV

D. (1) EPP-7, SI Termination --> EPP-27, Loss of DC Bus "B"

(2) Verify FCV-626 is open and then secure all charging pumps, control pressurizer pressure using control and backup heaters and normal spray flow

The correct answer is A.

A. Correct. The indications given indicate that a loss of DC Bus "B" has occurred. When on normal plant electric plant lineup, a safety injection will not occur just due a loss of DC Bus "B". EPP-4 will be performed until it directs the crew to transition to EPP-7. EPP-27 will direct the crew to isolate letdown, reduce charging flow and control pressure using heaters and PORV. Spray will be lost due to IA being isolated to the CV on a loss of DC Bus "B".

B. Incorrect - The first half of answer is correct. The second half is incorrect. Isolation of letdown has been omitted. Charging pumps will only be secured once PZR is greater than 71%. Also, normal spray flow is not available since PCV-1716 has failed closed isolating instrument air to containment.

C. Incorrect - The first half is incorrect. A Safety Injection signal is not received on a loss of DC Bus "B" from a normal electric plant lineup. EPP-7, SI Termination, is an entry condition to EPP-27 if an SI had occurred. The second part of the distractor is correct.

D. Incorrect. The first half is incorrect. A Safety Injection signal is not received on a loss of DC Bus "B" from a normal electric plant lineup. EPP-7, SI Termination, is an entry condition to EPP-27 if an SI had occurred. The second half is incorrect. Isolation of letdown has been omitted. Charging pumps will only be secured once PZR is greater than 71%. Also, normal spray flow is not available since PCV-1716 has failed closed isolating instrument air to containment.

Question 79
Tier/Group 1/1
K/A Importance Rating - RO 3.5 SRO 3.9

Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems

Reference(s) - Sim/Plant design, EPP-4/7, EPP-27, Foldout H
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-27-002
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments -

SRO: Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. Candidate must also demonstrate knowledge of the content of the procedure. ?

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Moisture Seperator Reheater Steam Valves - CLOSED** • MSR Shutoff Valves • MSR Purge Valves	<u>IF</u> a loss of power prevents isolation of the MSRs, <u>THEN</u> close the MSIVs <u>AND</u> MSIV BYPs. <u>IF</u> ANY Purge <u>OR</u> Shutoff Valve can <u>NOT</u> be closed from the RTGB <u>AND</u> RCS temperature is less than 540°F and lowering, <u>THEN</u> close the MSIVs <u>AND</u> MSIV BYPs. Locally close Open MSR Steam Valves
* 2.	**Determine If Procedure Exit Is Warranted:**	
	a. Check Attack on RNP Site - IN PROGRESS	a. Go To Step 3
	b. Check either of the below events - IN PROGRESS • Total Loss Of SW <u>OR</u> • Loss Of Lake Robinson Dam integrity	b. <u>IF</u> a total loss SW <u>OR</u> a loss of Lake Robinson Dam integrity occurs due to hostile action, <u>THEN</u> Go To EPP-28, Loss of Ultimate Heat Sink. Go To Step 3.
	c. Go To EPP-28, Loss Of Ultimate Heat Sink	
* 3.	**Check SI Signal - INITIATED**	<u>IF</u> SI initiation occurs during this procedure, <u>THEN</u> Go To Path-1, Entry Point A. Go To Step 5.
4.	**Go To Path-1, Entry Point A**	
5.	**Perform The Following:**	
	a. Reset SPDS	
	b. Initiate monitoring of Critical Safety Function Status Trees	
6.	**Open Foldout H**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*25.	**Control S/G Levels As Follows:**	
	a. Check S/G levels - ANY GREATER THAN 8%	a. Perform the following: 1) Establish FW Flow using one of the following: • Establish FW bypass flow greater than $0.2x10^6$ pph. <u>OR</u> • Establish AFW flow greater than 300 gpm. 2) <u>WHEN</u> S/G level is greater than 8%, <u>THEN</u> perform Step 25.b. 3) Go To Step 26.
	b. Control feed flow to maintain S/G levels - BETWEEN 39% <u>AND</u> 50%	
*26.	**Check DC Busses "A" <u>AND</u> "B" - ENERGIZED**	Go To the appropriate procedure: • DC Bus "A" - EPP-26, LOSS OF DC BUS "A" • DC Bus "B" - EPP-27, LOSS OF DC BUS "B"
*27.	**Check All The Following EDG Cooling Annunciators - EXTINGUISHED** • APP-010-E2, EDG A LUBE OIL HI/LO TEMP • APP-010-E3, EDG B LUBE OIL HI/LO TEMP • APP-010-F2, EDG A COOL WTR HI/LO TEMP • APP-010-F3, EDG B COOL WTR HI/LO TEMP	Perform the following: a. Locally determine if a high temperature condition exists. b. <u>IF</u> a high temperature condition exists, <u>THEN</u> shutdown the affected EDG.

FOLDOUT H

(Page 1 of 4)

1. 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%]

FOLDOUT H

(Page 2 of 4)

2. DC BUS OR INSTRUMENT BUS FAILURE CRITERIA

a. IF DC Bus failure has occurred, THEN perform the following:

1) IF DC Bus A fails, THEN perform the following:

a) In the Charging Pump Room, Open CVC-358, RWST TO CHARGING PUMP SUCTION.

b) WHEN CVC-358 is open, THEN close LCV-115C, VCT OUTLET from RTGB.

c) In the E-1/E-2 Room, transfer Instrument Bus 2 to MCC-8.

d) In the 4160V Bus Room, trip the Exciter Field Breaker.

e) In EDG A Room perform the following:

- Trip EDG A Fuel Racks.
- Close DA-21A AND DA-25A, DG "A" AIR START OUTLET ISOLATION valves.

2) IF DC Bus B fails, THEN perform the following:

a) In the E-1/E-2 Room, transfer Instrument Bus 3 to MCC-8.

b) In EDG B Room, perform the following:

- Trip EDG B Fuel Racks.
- Close DA-21B AND DA-25B, DG "B" AIR START OUTLET ISOLATION valves.

c) Close LCV-460 A & B, LTDN LINE STOPs.

b. IF MCC-5 is de-energized, THEN transfer power source to DS Bus using the posted instructions at the Kirk Key Interlocked Breakers.

(CONTINUED NEXT PAGE)

FOLDOUT H

(Page 3 of 4)

2. (CONTINUED)

   c. IF Instrument Bus failure has occurred, THEN perform the following:

      1) IF Instrument Bus 4 fails, THEN maintain Steam Dump in the Tavg Mode of operation.

      2) IF a failure of only ONE of the below Instrument Busses occurs, THEN transfer the failed bus to MCC-8.

         - Instrument Bus 1
         - Instrument Bus 2
         - Instrument Bus 3
         - Instrument Bus 4

      3) IF more than ONE Instrument Bus requires transfer to MCC-8 for Nuclear Safety Concerns, THEN strip the affected Busses using Attachment 13 of AOP-024, Loss of Instrument Bus, prior to transferring the Buss(es) to MCC-8.

3. LOSS OF RCP SEAL COOLING CRITERIA
   IF both the conditions below are met, THEN perform AOP-018, Reactor Coolant Pump Abnormal Conditions to restore RCP Seal Cooling:

   - APP-001-B2, LABYRINTH SEAL LOW ΔP - ILLUMINATED

     AND

   - APP-001-D1, THERMAL BARRIER LO FLOW - ILLUMINATED

FOLDOUT H

(Page 4 of 4)

4. EMERGENCY COOLING WATER SWITCHOVER CRITERIA

IF normal cooling is lost to any of the following components, THEN establish emergency cooling water using the referenced procedure:

- Charging Pump Oil Coolers - Use Attachment 1 of AOP-014, Component Cooling Water System Malfunction.
- MDAFW Pumps - Use Attachment 2 of AOP-022, Loss of Service Water.

5. AFW SUPPLY SWITCHOVER CRITERIA

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

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Electrical Status At Time DC Power Was Lost - UNIT SYNCHRONIZED TO THE GRID	Go To Step 3.
2.	Inform Load Dispatcher Of The Following: a. A loss of DC Control Power has occurred b. Switchyard OCBs 52/8 <u>AND</u> 52/9 have failed to trip c. Backup relaying has caused a North Bus Lock Out which tripped the following Switchyard Circuit Breakers: • 52/3, ROCKINGHAM 230KV • 52/6, #1 230-115KV BANK 230KV • 52/7, DARLINGTON SCPSA 230KV • 52/12, DARLINGTON COUNTY PLANT SOUTH 230KV • 52/14, DARLINGTON COUNTY PLANT NORTH 230KV d. Request that Load Dispatcher send a Maintenance Crew to locally trip OCBs 52/8 <u>AND</u> 52/9	
* 3.	Check The Cause Of The DC Bus Failure - KNOWN	<u>WHEN</u> the cause is determined, <u>THEN</u> notify Maintenance to correct the problem. Observe the <u>NOTE</u> prior to Step 4 and Go To Step 4.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

AFW PUMP B will not be available due to loss of control power to the breaker.

4. Maintain S/G Levels Between 8% And 50% Using Available AFW Pumps:

   - AFW PUMP A
   - STEAM DRIVEN AFW PUMP

5. Close LCV-460A & B, LTDN LINE STOP

6. Locally Perform Attachment 2 To Restore Power To 4160V <u>AND</u> 480V Busses While Continuing With This Procedure

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

FCV-626 fails closed on a loss of DC Bus B. Seal injection flow is required for RCP Seal Cooling.

****

NOTE

Instrument Air to the CV is isolated (IA PCV-1716 fails closed). This results in a loss of all PZR Spray AND Letdown. The following steps will control PZR level and pressure.

7. Lower Charging Flow To Minimum As Follows:

   a. Stop all but one Charging Pump

   b. Verify the running Charging Pump Speed Controller in AUTO:

      - Pump A: SC-151
      - Pump B: SC-152
      - Pump C: SC-153

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
8.	Check RCP Seal Injection Flow - BETWEEN 8 GPM AND 13 GPM	Locally throttle RCP SEAL WATER FLOW CONTROL VALVE(s) to obtain flow to each RCP between 8 gpm and 13 gpm. • CVC-297A • CVC-297B • CVC-297C IF required to maintain minimum flow, THEN throttle HIC-121, CHARGING FLOW Valve while maintaining Charging Pump Discharge pressure less than 2500 PSIG. IF the normal Seal Injection Range can NOT be maintained, THEN an expanded range of between 6 gpm and 20 gpm may be used.
9.	Verify The Following: a. PCV-456, PZR PORV - IN AUTO b. RC-535, PORV BLOCK - OPEN	
*10.	Maintain PZR Pressure At Desired Pressure As Follows: a. Raise pressure as necessary using Control AND Backup Heaters	
	b. Check PZR Pressure - LESS THAN 2335 PSIG	b. Verify PCV-456, PZR PORV, opens to maintain pressure less than 2335 psig. WHEN pressure is less than 2335 psig, THEN verify the PORV recloses.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Starting duty limitations allow only 4 Charging Pump starts per hour and require a minimum of 5 minutes between starts.

****

*11.	Check PZR Level - GREATER THAN 71%	WHEN PZR level Is Greater Than 71%, THEN observe the NOTE prior to Step 12 and perform Steps 12 and 13 Go To Step 14.

NOTE

FCV-626 will have to be manually opened due to failing closed on a loss of DC Bus B. Seal injection flow is required for RCP Seal Cooling.

12.	Check APP-001-D1, RCP THERM BAR COOL WTR LO FLOW - EXTINGUISHED	Perform the following: a. Verify at least one CCW Pump RUNNING. b. Verify the following CCW valves are OPEN: • CC-716A, CCW TO RCP ISO • CC-716B, CCW TO RCP ISO • CC-735, THERM BAR OUT ISO • CC-730, BRG OUTLET ISO • FCV-626, THERM BAR FLOW CONT

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	Reduce SW Pressure As Follows:	
	a. Check number of SW Pumps Running - GREATER THAN 2	a. <u>WHEN</u> personnel are available, <u>THEN</u> locally perform Attachment 3, Throttling CCW Heat Exchanger SW Valves while continuing with this procedure. Go To Step 5.
	b. Stop 1 Pump	
	c. Check SW Header Pressure - GREATER THAN 50 PSIG	c. Go To Step 4.e.
	d. Go To Step 4.a	
	e. Check SW Header pressure - GREATER THAN 40 PSIG	e. <u>WHEN</u> personnel are available, <u>THEN</u> locally perform Attachment 3, Throttling CCW Heat Exchanger SW Valves while continuing with this procedure. Go To Step 5.
* 5.	Check DC Busses A <u>AND</u> B - ENERGIZED	Go To the appropriate procedure: • DC Bus A - EPP-26, LOSS OF DC BUS "A" • DC Bus B - EPP-27, LOSS OF DC BUS "B"

Distractor

80. W/E 04 EG2.4.3 SRO 001

The Plant has experienced a LOCA Outside of Containment and the crew has implemented PATH-1 and transitioned to EPP-20, LOCA Outside Containment.

Which ONE (1) of the following completes the statement below?

From the instruments listed below, __(1)__ are the Reg. Guide 1.97 (PAM) instruments **that will be utilized by EPP-20 to diagnose and mitigate** the accident and the conditions of ITS LCO 3.3.3, Post Accident Monitoring (PAM) Instrumentation are __(2)__.

PI-455, PZR Pressure	- 1750 psig and lowering
PI-456, PZR Pressure	- 1725 psig and lowering
PI-457, PZR Pressure	- Failed LOW
PI-511AA, RCS Wide Range Pressure	- 1750 psig and lowering
PI-511BA, RCS Wide Range Pressure	- 1750 psig and lowering
PI-402, RCS Wide Range Pressure	- 1725 psig and lowering
PI-501, RCS Wide Range Pressure	- Failed LOW
LT-459, PZR Level Transmitter	- 5% and lowering
LT-460, PZR Level Transmitter	- 5% and lowering
LT-461, PZR Level Transmitter	- 5% and lowering

(REFERENCE PROVIDED)

A. (1) PI-455, 456, 457, 402, 501, LT-459, 460, 461 (**NOT** PI-511AA, 511BA)

(2) NOT met

B. (1) PI-455, 456, 457, 402, 501 (**NOT** LT-459, 460 , 461, PI-511AA, 511BA)

(2) NOT met

C.✓ (1) PI-511AA, 511BA, 402, 501 (**NOT** LT-459, 460 , 461, PI-455, 456, 457)

(2) met

D. (1) PI-511AA, 511BA, LT-459, 460 , 461 (**NOT** PI-455, 456, 457, 402, 501)

(2) met

The correct answer is C.

A. Incorrect. PI-455,456,457 are not Reg. Guide 1.97 instruments. EPP-20 only uses pressure indication only as a means of diagnosis and mitigation for LOCA Outside Containment. As sections of systems are isolated RCS pressure is monitored to see if a rising trend is observed. Plausible if the candidate thinks that PI-402 and PI-501, which are Reg Guide 1.97, are the instruments credited for in ITS 3.3.3. The bases for ITS 3.3.3 states that the RCS Pressure (Wide Range) instruments are the indications from the Inadequate Core Cooling Monitor. PI-402 and PI-501 are not part of the ICCM and are therefore not credited in ITS 3.3.3.

B. Incorrect. See discussion A above.

C. Correct. PI-511AA, 511BA, 402 and 501 are the only Reg. Guide 1.97 instruments listed in the stem of the question. EPP-020 only uses RCS pressure as an indication to determine if actions to isolate the LOCA have been successful. ITS LCO 3.3.3 requires that 2 RCS Wide Range Pressure instruments be operable. The bases for ITS 3.3.3 states that the RCS Pressure (Wide Range) instruments are the indications from the Inadequate Core Cooling Monitor. PI-402 and PI-501 are not part of the ICCM and are therefore not credited in ITS 3.3.3.

D. Incorrect. See distractor A discussion. The second half of the distractor is correct.

Question 80
Tier/Group 1/1
K/A Importance Rating - RO 3.7 SRO 3.9

LOCA Outside Containment: Ability to identify post-accident instrumentation.

Reference(s) - Sim/Plant design, EPP-20, OMM-007, TMM-026, ITS LCO 3.3.3 and bases.
Proposed References to be provided to applicants during examination - ITS 3.3.3
Learning Objective -EPP-20-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.6 / 45.4
Comments - K/A match because candidate must know which instruments listed are utilized in EPP-20 and what the requirements are for PAM instrumentation operability.

SRO: Candidate must demonstrate knowledge of TS bases that is required to analyze TS required actions and terminology. The TS bases is the only document that specifies that the Wide Range Pressure indication on the ICCM are the ITS 3.3.3 credited RCS pressure indications.

81. W/E 05 EA2.1 SRO 001

Given the following plant conditions:

- Plant is operating at 100% RTP.
- A feedwater break occurs inside containment.
- Containment pressure reached 4.2 psig and is currently 3.5 psig and lowering.
- AFW flow CANNOT be established.
- The crew has transitioned from PATH-1 to FRP-H.1, Response to Loss of Secondary Heat Sink.
- S/G conditions:

S/G	Level (WR)
A	10%
B	19%
C	18%

Which ONE (1) of the following completes the statement below?

Transition to the RCS Bleed and Feed steps of FRP-H.1 __(1)__ required.
The **NEXT** action required to be performed by the crew will be to __(2)__.

A. (1) is

(2) initiate a safety injection signal

B.✓ (1) is

(2) stop all RCPs

C. (1) is NOT

(2) verify all S/G blowdown and sample isolation valves - CLOSED

D. (1) is NOT

(2) check that the AFW lines are intact

The correct answer is B.

A. Incorrect. The first part is correct. The RCPs are to be secured immediately due to a loss of heat sink. Initiation of a safety injection signal is the second action the crew will take when establishing bleed and feed.

B. Correct.

C. Incorrect. Adverse numbers exist. Transition to bleed and feed are any two S/G wide range levels less than 10% [19%]. Two steam generators are less than 19%, therefore bleed and feed is required. The second half would be correct had bleed and feed not been required.

D. Incorrect. Adverse numbers exist. Transition to bleed and feed are any two S/G wide range levels less than 10% [19%]. Two steam generators are less than 19%, therefore bleed and feed is required. The second half would have been the second action the crew would have taken had bleed and feed not been required.

Question 81
Tier/Group 1/1
K/A Importance Rating - RO 3.4 SRO 4.4

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 operations.

Reference(s) - Sim/Plant design, FRP-H.1, OMM-022
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-H.1-001, -004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13

SRO: The candidate must assess the given plant conditions during the emergency given and the determine is a given section is applicable. This is equivalent to the knowledge of diagnostic steps and decision points in the EOP network that involves transitions to event specific sub-procedures or emergency contingency procedures.

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 10% [19%] yes	IF any two S/G Wide Range Levels lower to less than 10% [19%], THEN Go To Step 5. Go To Step 6.
5.	Perform The Following: a. Stop all RCPs b. Observe CAUTION prior to Step 31 and Go To Step 31	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
29.	Determine If Condensate Flow Is Adequate:	
	a. Check the following: • Core Exit T/C temperature - LOWERING OR • S/G Wide Range level - RISING IN AT LEAST ONE S/G	a. Go To Step 30.
	b. Maintain FW flow to restore S/G level to greater than 8% [18%]	
	c. Reset SPDS And Return To Procedure And Step In Effect	
*30.	Any Two S/G Wide Range Levels - LESS THAN 10% [19%]	IF FW flow is restored during steps prior to step 31, THEN Go To Step 28. Go To Step 3.

****

CAUTION

Steps 31 through 35 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
31.	Depress the INITIATE SAFETY INJECTION Pushbutton	DISTRACTOR.

8.3.13 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.

8.3.14 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.

8.3.15 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.

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 SM 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".

82. W/E 015 EG2.1.32 SRO 001

Given the following plant conditions:

- A seismic event has lead to a LOCA and rupture of SW piping in the CV.
- PATH-1 completed to the point of transition to another procedure.
- Only one train of CV Spray is operating properly.
- The following conditions exist in Containment:
  - Containment Pressure is 24 psig and rising.
  - Containment Sump Level is 385 inches and rising.

Which ONE (1) of the following completes the statement below?

Based on the current conditions, containment __(1)__ is the highest priority and the basis for this is __(2)__.

A. (1) pressure

(2) one train of CV Spray cannot maintain pressure below the containment design pressure

B. (1) pressure

(2) one train of CV Spray along with loss of SW in containment cannot maintain pressure below the containment design values

C✓ (1) level

(2) potential flooding of critical systems and components needed for future plant recovery may occur

D. (1) level

(2) dilution of sump water with SW may potentially cause a return to criticality

The correct answer is C.

A. Incorrect. As long as one train of CV spray is operating containment pressure will not become the top priority until pressure reaches 42 psig. Since above 10 psig, the operator is sent to a section in the CSFSTs that verifies adequate spray flow.

B. Incorrect. As long as one train of CV spray is operating containment pressure will not become the top priority until pressure reaches 42 psig. Since above 10 psig, the operator is sent to a section in the CSFSTs that verifies adequate spray flow.

C. Correct.

D. Incorrect. The first half of distractor is correct. The basis for FRP-J.2, Containment Flooding, does NOT address reactivity as a concern. The rupture of SW will cause a dilution of the CV Sump water.

Question 82
Tier/Group 1/1
K/A Importance Rating - RO 3.8 SRO 4.0

Containment Flooding: Ability to explain and apply system limits and precautions.

Reference(s) - Sim/Plant design, CSFST, FRP-J.1 / J.2, FRP-J.2BD
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-J.2-002, --3
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.2 / 45.12

SRO: Candidate must know the hierarchy of the Critical Safety Function Status Trees with respect to containment parameters. Candidate must also know the bases for this functional restoration procedure.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3
PART 4

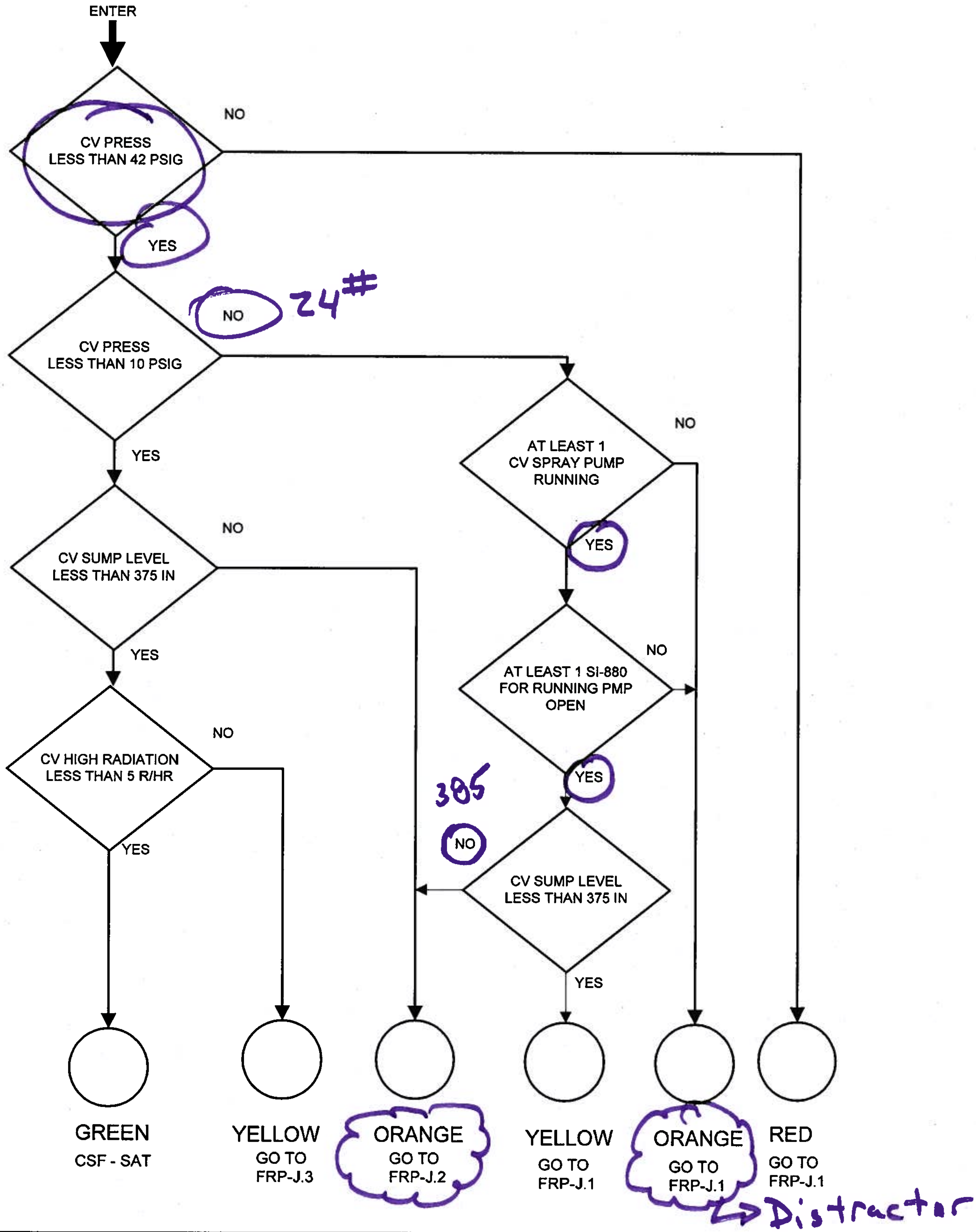
Emergency Operating Procedure

CSFST

CRITICAL SAFETY FUNCTION STATUS TREES

REVISION 4

CSF-5, CONTAINMENT



CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

FUNCTION RESTORATION PROCEDURE

FRP-J.1

RESPONSE TO HIGH CONTAINMENT PRESSURE

REVISION 8

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to respond to high containment pressure.

2. ENTRY CONDITIONS

   CSF-5, Containment Critical Safety Function Status Tree on a RED, ORANGE or YELLOW condition.

- END -

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. <u>IF</u> any CONTAINMENT ISOLATION PHASE A Valve fails to close, <u>THEN</u> 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. <u>IF</u> any CONTAINMENT VENTILATION ISOLATION Valve fails to close, <u>THEN</u> locally isolate the affected penetration.
3.	Check CV Pressure - HAS RISEN 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 7	
* 5.	Check Spray Additive Tank Level - GREATER THAN 0%	Verify Spray Additive Tank isolated as follows: • SI-845A, SAT DISCH. CLOSED • SI-845B, SAT DISCH. CLOSED • SI-845C, SAT THROTTLE VALVE, CLOSED

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
6.	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.
7.	Verify CONTAINMENT ISOLATION PHASE B Valves - CLOSED	
8.	Verify All RCPs - STOPPED	
9.	Verify CV AIR RECIRC COOLERs - RUNNING • HVH-1 • HVH-2 • HVH-3 • HVH-4	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
10.	Verify The Following: a. All MSIV AND MSIV BYP Valves - CLOSED b. All MSIV Control Switches - SELECTED TO CLOSE	
11.	Check S/G Status - AT LEAST ONE INTACT	Perform the following: a. Maintain feed flow to each S/G between 80 gpm and 90 gpm. b. Go To Step 13.

****

CAUTION

At least one S/G must be maintained available for RCS cooldown.

****

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
12.	Determine If Any S/G Is Faulted As Follows:	
	a. Check pressures in all S/Gs: • ANY S/G PRESSURE LOWERING IN AN UNCONTROLLED MANNER OR • ANY S/G COMPLETELY DEPRESSURIZED	a. Go To Step 13.
	b. Isolate affected S/G(s) using EPP-SUPPLEMENTS, Supplement G, S/G Isolation	
13.	Reset SPDS AND Return To Procedure And Step In Effect	

- END -

1A

Continuous Action Steps

(Page 1 of 1)

Main Body

5. Check Spray Additive Tank Level - GREATER THAN 0%.
<u>WHEN</u> tank level reaches 0%, <u>THEN</u> isolate tank.

1B

Continuous Action Steps

(Page 1 of 1)

Main Body

5. Check Spray Additive Tank Level - GREATER THAN 0%.
WHEN tank level reaches 0%, THEN isolate tank.

CONTINUOUS USE

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON SEG PLANT

PLANT OPERATING MANUAL

VOLUME 3

PART 4

FUNCTION RESTORATION PROCEDURE

FRP-J.2

RESPONSE TO CONTAINMENT FLOODING

REVISION 3

Effective Date 12/16/94

RECOMMENDED BY: Gary M Smiley for 8/1/94
Operations Procedure Coordinator — Date

APPROVED BY: CMW for 8/1/94
Manager - Operations — Date

1.0 PURPOSE

This procedure provides actions to respond to containment flooding.

2.0 ENTRY CONDITIONS

CSF-5, Containment Critical Safety Function Status Tree on an ORANGE condition.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Any S/G - FAULTED <u>OR</u> RUPTURED	Go To Step 3.
2.	Verify Any Faulted <u>OR</u> Ruptured S/G - ISOLATED USING SUPPLEMENT G	
3.	Determine If A CCW Leak Exists As Follows: • Check CCW Surge Tank level - DECREASING <u>OR</u> • Check excessive makeup required to maintain CCW Surge Tank Level stable	Go To Step 5.
4.	Isolate CCW To Containment As Follows: a. Verify All RCPs - STOPPED b. Locally close CC-737A, CCW SUPPLY TO EXCESS LETDOWN HEAT EXCHANGER Valve c. Verify the following CCW Containment Isolation Valves - CLOSED: • CC-739, CCW FROM EXCESS LTDN HX • CC-716A, CCW TO RCP ISO • CC-716B, CCW TO RCP ISO • CC-626, THERMAL BAR FLOW CONT • CC-730, BRG OUTLET ISO • CC-735, THERMAL BAR OUT ISO	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Determine If A Fire Water Leak Exists As Follows: • Check APP-044-C55, E.D. FIRE PUMP RUNNING - CLEAR <u>AND</u> • Check APP-044-C58, M.D. FIRE PUMP RUNNING - CLEAR	Verify The Following Fire Water Containment Isolation Valves - CLOSED: • FP-256, RCP SPRINKLER ISOLATION VALVE • FP-258, RCP SPRINKLER ISOLATION VALVE • FP-248, ELECT PENETRATION CV ISOLATION • FP-249, ELECT PENETRATION CV ISOLATION
6.	Verify The Following Primary Water Containment Isolation Valves - CLOSED: • RC-519A, PW TO CV ISO • RC-519B, PW TO CV ISO	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Perform The Following To Identify An HVH Cooler Leak:	
	a. Check APP-002-A8, HVH-1 WTR OUTLET FLOW LO Alarm - EXTINGUISHED	a. Perform the following to isolate SW flow to HVH-1: 1) Verify HVH-1, CV RECIRC FAN is STOPPED. 2) Verify the following valves are CLOSED: • V6-33A, SW INLET • V6-34A, SW OUTLET • V6-35A, WTR SAMPLING
	b. Check APP-002-B8, HVH-2 WTR OUTLET FLOW LO Alarm - EXTINGUISHED	b. Perform the following to isolate SW flow to HVH-2: 1) Verify HVH-2, CV RECIRC FAN is STOPPED. 2) Verify the following valves are CLOSED: • V6-33B, SW INLET • V6-34B, SW OUTLET • V6-35B, WTR SAMPLING • V6-33F, SELECTIVE INLET
	c. Check APP-002-C8, HVH-3 WTR OUTLET FLOW LO Alarm - EXTINGUISHED	c. Perform the following to isolate SW flow to HVH-3: 1) Verify HVH-3, CV RECIRC FAN is STOPPED. 2) Verify the following valves are CLOSED: • V6-33C, SW INLET • V6-34C, SW OUTLET • V6-35C, WTR SAMPLING

(CONTINUED NEXT PAGE)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	(CONTINUED)	
	d. Check APP-002-D8, HVH-4 WTR OUTLET FLOW LO Alarm - EXTINGUISHED	d. Perform the following to isolate SW flow to HVH-4: 1) Verify HVH-4, CV RECIRC FAN is STOPPED. 2) Verify the following valves are CLOSED: • V6-33D, SW INLET • V6-34D, SW OUTLET • V6-35D, WTR SAMPLING • V6-33E, SELECTIVE INLET
8.	Notify Chemistry To Sample The RHR System To Determine CV Sump Water Activity	Contact Plant Operations Staff to determine method for sampling CV Sump.
9.	Notify Plant Operations Staff Of Containment Water Level AND Activity Level To Obtain Recommended Actions	
10.	Reset SPDS AND Return To Procedure And Step In Effect	

- END -

83. 037 AG2.4.41 SRO 001

Given the following plant conditions:

- The plant is at 100% RTP.
- A tube leak is identified in "C" S/G.
- Two charging pumps running at maximum speed are able to control and stabilize PZR level with one 45 gpm letdown orifice in service.
- Total RCP Seal Leakoff flow is 7.5 gpm.

Which ONE (1) of the following completes the statement below?

The required action to address this condition is to __(1)__ and EAL Classification for this event is __(2)__.

AOP-035, S/G Tube Leak
GP-006-1, Normal Plant Shutdown from Power Operation to Hot Shutdown.

(REFERENCE PROVIDED)

A. (1) Shutdown the plant IAW GP-006-1 while concurrently performing AOP-035

(2) Unusual Event SU6.1, RCS leakage

B.✓ (1) Shutdown the plant IAW GP-006-1 while concurrently performing AOP-035

(2) Alert FA1.1, Any loss or any potential loss of either Fuel Clad or RCS

C. (1) Isolate "C" S/G IAW AOP-035 and then commence a plant shutdown IAW GP-006-1

(2) Unusual Event SU6.1, RCS leakage

D. (1) Isolate "C" S/G IAW AOP-035 and then commence a plant shutdown IAW GP-006-1

(2) Alert FA1.1, Any loss or any potential loss of either Fuel Clad or RCS

The correct answer is B.

A. Incorrect. The first half of distractor is correct. The EAL classification is incorrect because the leak rate given is greater than the capacity of one charging pump. A UE would be valid up to primary to secondary leak rate of 77 gpm.

B. Correct.

C. Incorrect. Isolation of the S/G is not a prerequisite to commencing the plant shutdown. However in some cases the crew may progress through AOP-035 S/G isolation prior to actually starting the shutdown. The EAL classification is incorrect because the leak rate given is greater than the capacity of one charging pump. A UE would be valid up to primary to secondary leak rate of 77 gpm.

D. Incorrect. Isolation of the S/G is not a prerequisite to commencing the plant shutdown. However in some cases the crew may progress through AOP-035 S/G isolation prior to actually starting the shutdown. The second half of the distractor is correct.

Question 83
Tier/Group 1/2
K/A Importance Rating - RO 2.9 SRO 4.6

Steam Generator Tube Leak: Knowledge of the emergency action level thresholds and classifications.

Reference(s) - Sim/Plant design,
Proposed References to be provided to applicants during examination - EAL Matrix
Learning Objective - AOP-035-004, EAL 004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.11
Comments -

SRO: Candidate must know the proper procedure progression for the given condition and requires that an EAL classification be properly determined.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
5.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
6.	Adjust Charging Flow As Follows:	
	a. Check Charging Pump Status - AT LEAST TWO RUNNING	a. Start one additional Charging Pump.
	b. Place running Charging Pumps Speed Controllers in MAN AND adjust output to maximum	
7.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
8.	Check Letdown - IN SERVICE	Go To Step 11.
9.	Verify All Letdown Flowpaths Isolated As Follows: • LCV-460A & B, LTDN LINE STOP Valves - CLOSED • HIC-137, EXCESS LTDN FLOW Controller - ADJUSTED TO 0% • CVC-387, EXCESS LTDN STOP - CLOSED	
10.	Check RCS Level - LOWERING IN AN UNCONTROLLED MANNER	Go To Step 12.
11.	Trip The Reactor AND Go To Path-1 OR EOP-E-0, Reactor Trip or Safety Injection	
12.	Control Charging Flow To Maintain Desired RCS Level	
13.	Check RCS Leakage - GREATER THAN RUNNING CHARGING FLOW	IF leakage exceeds Charging flow, THEN Go To Step 6. Go To Step 15.
14.	Go To Step 6	
15.	Notify Chemistry Personnel To Periodically Sample All S/Gs For Activity And Boron Concentration	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*16.	Check Assistance To Open S/G Sample Valves- REQUESTED	WHEN assistance to open S/G Sample valves is requested, THEN observe the NOTE prior to Step 17 and perform Step 17 Observe the NOTE prior to Step 18 and Go To Step 18

NOTE

Operation of the S/G Blowdown OVERRIDE OPEN key switches will result in an ITS 3.6.3 entry.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
17.	WHEN Requested By Chemistry Personnel To Support S/G Samples, THEN Perform The Following:	
	a. Check R-19 Monitor - IN ALARM OR EXPECTED TO ALARM	a. Observe the NOTE prior to Step 18 and Go To Step 18
	1) Place the S/G Blowdown OVERRIDE OPEN key switch for the affected S/G(s) in the OVERRIDE OPEN position • S/G A BLOWDOWN SGB-1933A & SGB-1933B • S/G B BLOWDOWN SGB-1934A & SGB-1934B • S/G C BLOWDOWN SGB-1935A & SGB-1935B	
	2) Within ONE Hour restore the S/G Blowdown OVERRIDE OPEN key switch for the affected S/G(s) to the NORMAL position (ITS 3.6.3. Condition B)	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Radiation Monitor R-24 does not provide an accurate determination of leakage until S/G samples have been obtained and the monitor has been calibrated for the optimal node for leakage location.

18. Determine Leak Rate Using At Least One Of The Following Methods:

- R-24 Recorder
- Perform OST-051, Reactor Coolant System Leakage Evaluation
- Perform a Charging versus Letdown balance
- Notify Chemistry personnel to perform isotopic analysis of S/G samples for leak rate determination
- Use R-15 to monitor for low level Primary-to-Secondary leakage using the OP-504, Condenser Air Removal section "Using R-15 to Monitor for Low Level Primary to Secondary Leakage"
- Use CP-014 Conversion Factors to correlate R-15 to leakage

19. Check Leak Rate Determination - COMPLETE

RESPONSE NOT OBTAINED: WHEN leak rate determination is complete, THEN observe the NOTE prior to Step 20 and Go To Step 20.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

- ITS LCO 3.4.13 provides a primary to secondary leakage limit of 75 gpd through any one S/G.
- Total leakage is assumed to be coming from a single S/G when unable to determine leakage from the individual S/Gs.
- Normally performed steps in GP-006-1 or GP-006-2, such as placing S/G Blowdown to the Flash Tank may require Release Permits.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
*20.	Check Leak Rate - GREATER THAN <u>OR</u> EQUAL TO 100 GPD FOR A SINGLE S/G	<u>IF</u> the leak rate exceeds the limit, <u>THEN</u> Go To Step 21. Go To Step 22.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

It is important to perform GP-006-1 or GP-006-2 and AOP-035 concurrently to the extent possible in order to minimize secondary contamination.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
21.	Perform The Following Power Reduction:	
	a. Notify Chemistry that a PSAL-3 event has occurred	
	b. Check Reactor Status - MODE 1 OR MODE 2	b. Observe the NOTE prior to Step 24 and Go To Step 24
	c. Initiate Plant Shutdown To Mode 3 Using GP-006-1, Normal Plant Shutdown From Power Operation To Hot Shutdown, OR GP-006-2, Rapid Plant Shutdown, While Continuing With This Procedure	
	d. Adhere to the following time limits: • Be less than 50% power within 1 hour of declaring PSAL-3 • Be in Mode 3 within 3 hours of declaring PSAL-3	
	e. Observe the NOTE prior to Step 24 and Go To Step 24	
*22.	Check Leak Rate - GREATER THAN OR EQUAL TO 75 GPD FOR A SINGLE S/G	IF the leak rate exceeds the limit, THEN Go To Step 23. Go To Step 105.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

It is important to perform GP-006-1 OR GP-006-2 and AOP-035 concurrently to the extent possible in order to minimize secondary contamination.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
23.	Perform The Following Power Reduction:	
	a. Notify Chemistry that a PSAL-2 event has occurred	
	b. Check Reactor Status - MODE 1 OR MODE 2	b. Observe the NOTE prior to Step 24 and Go To Step 24
	c. Initiate Plant Shutdown To Mode 3 Using GP-006-1, Normal Plant Shutdown From Power Operation To Hot Shutdown, OR GP-006-2, Rapid Plant Shutdown, While Continuing With This Procedure	
	d. Be in Mode 3 within 6 hours of declaring PSAL-2	
	e. Observe the NOTE prior to Step 24 and Go To Step 24	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

Radiation Monitor R-24 does not provide an accurate determination of leakage until S/G samples have been obtained and the monitor has been calibrated for the optimal node for leakage location.

24. Identify Leaking S/G Using At Least One Of The Following Methods:

- Evaluate indications on R-24 Recorder

  OR

- Evaluate indications on RI-19A, RI-19B, and RI-19C, STM GEN BLOW DN Radiation Monitors

  OR

- Evaluate indications on R-31A, R-31B, and R-31C, STEAMLINE RADIATION MONITORs

  OR

- Chemistry analysis of S/G samples for boron and activity

25. Implement The EALs

26. Review Technical Specification LCOs

- ITS LCO 3.4.13
- ITS LCO 3.7.15
- ITS LCO 3.6.3

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
27.	Initiate Monitoring RCS Leak Rate As Follows:	
	a. Check Radiation Monitor R-24 - IN SERVICE	a. Go To Step 27.d.
	b. Log R-24 readings at 15 minute intervals	
	c. Go To Step 28	
	d. Check RCS Leak Rate - LESS THAN 10 GPM	d. Log RCS leakage estimates at 15 minute intervals. Go To Step 28.
	e. Monitor R-15 Trends Using Attachment 5, R-15 Monitoring	
28.	Contact An Operator To Bypass The Condensate Polishers As Follows:	
	a. Place the SECONDARY BYPASS Switch to the OPEN position	
	b. Depress the OFF pushbutton for each in service demineralizer	
29.	Perform Attachment 4, Controlling Secondary Contamination, While Continuing With This Procedure	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
30.	Isolate Non-essential Flowpaths From The Affected S/G As Follows:	
	a. Verify SDAFW Pump STEAM SHUTOFF from affected S/G - CLOSED: • V1-8A OR • V1-8B OR • V1-8C	
	b. Perform Attachment 1, Local S/G Isolation, while continuing with this procedure	S/G Isolation

NOTE

At high primary to secondary leak rates the ability to cooldown will be limited by RCS makeup capability. IF a rapid RCS cooldown is required AND difficulty is experienced maintaining PZR level, THEN consideration should be given to initiation of Safety Injection.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
31.	Check Turbine - TRIPPED	WHEN the Turbine is tripped as directed by GP-006-1 or GP-006-2, THEN Go To Step 32.
*32.	Check The Following Rod Banks - FULLY INSERTED: • All Control Banks • Shutdown Bank B • Shutdown Bank A	WHEN the rod banks are fully inserted, THEN perform Step 33. Go To Step 34.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

33.	Perform ONE Of The Following:
	• Verify Reactor Trips Breakers - OPEN
	OR
	• Withdraw The Following Rod Banks to 5 steps:
	  • Shutdown Bank A
	  • Shutdown Bank B
	  • Control Bank A
	  • Control Bank B
	  • Control Bank C
	  • Control Bank D	

34.	Control Charging Flow To Maintain PZR Level Between 22% and 53%	
35.	Borate The RCS Using Attachment 3, Establishing Mode 5 Boron Concentration, While Continuing With This Procedure	
36.	Check Tavg - LESS THAN 547°F	Perform one of the following to reduce Tavg between 543°F and 547°F: • Slowly adjust PC-464B, STEAM HEADER PRESS Controller auto setpoint pot OR • Slowly adjust PC-464B output in MAN <u>WHEN</u> Tavg is between 543°F and 547°F, <u>THEN</u> Go To Step 37.
37.	Perform Attachment 2, S/G Isolation At Hot Shutdown	

CONTINUOUS USE

ATTACHMENT 1

LOCAL S/G ISOLATION

NOTE

MS-20 is located in the pipe jungle approximately one foot above V1-8A.

1. IF S/G "A" is to be isolated, THEN perform the following:

   a. Close MS-20, SG "A" BYPASS DRN & WARM-UP LINE TO AFW PUMP.

   b. Verify CLOSED MS-19, SG "A" STEAM LINE BEFORE SEAT DRAIN ROOT ISOL.

   c. Verify CLOSED MS-21, SG "A" STEAM STOP V1-3A AFTER SEAT DRAIN ROOT ISOL.

   d. Close the S/G Blowdown Isolation and Sample Valves by removing power to R-19A (ON-OFF switch located inside the large cabinet, bottom right hand corner).

   e. Go To Step 4.

CONTINUOUS USE

ATTACHMENT 1

LOCAL S/G ISOLATION

(Page 2 of 3)

> **NOTE**
>
> MS-29 is located in the pipe jungle approximately one foot above V1-8B.

2. IF S/G "B" is to be isolated, THEN perform the following:

   a. Close MS-29, SG "B" BYPASS DRAIN & WARM-UP LINE TO AFW PUMP.

   b. Verify CLOSED MS-28, SG "B" STEAM LINE BEFORE SEAT DRAIN ROOT ISOL.

   c. Verify CLOSED MS-30, SG "B" STEAM STOP V1-3B AFTER SEAT DRAIN ROOT ISOL.

   d. Close the S/G Blowdown Isolation and Sample Valves by removing power to R-19B (ON-OFF switch located inside the large cabinet, bottom right hand corner).

   e. Go To Step 4.

> **NOTE**
>
> MS-38, is located in the pipe jungle approximately one foot above V1-8C.

3. IF S/G "C" is to be isolated, THEN perform the following:

   a. Close MS-38, SG "C" BYPASS DRAIN & WARM-UP LINE TO AFW PUMP.

   b. Verify CLOSED MS-37, SG "C" STEAM LINE BEFORE SEAT DRAIN ROOT ISOL.

   c. Verify CLOSED MS-39, SG "C" STEAM STOP V1-3C AFTER SEAT DRAIN ROOT ISOL.

   d. Close the S/G Blowdown Isolation and Sample Valves by removing power to R-19C (ON-OFF switch located inside the large cabinet, bottom right hand corner).

CONTINUOUS USE

ATTACHMENT 1

LOCAL S/G ISOLATION

(Page 3 of 3)

4. Notify Control Room personnel that actions of Attachment 1 to isolate the affected S/G are complete.

- END -

CONTINUOUS USE

ATTACHMENT 2

S/G ISOLATION AT HOT SHUTDOWN

(Page 1 of 3)

NOTE

- Closure of the MSIV in the affected S/G will invalidate the readings from Radiation Monitor R-15 for use in determining primary to secondary leak rate.
- Opening the breakers for the any V1-8, V2-14, OR V2-16 valve will require entry into Technical Specifications 3.7.4
- Based upon leak size and equipment availability, breakers may need to be re-closed to support maintaining level in the isolated S/G per this procedure.

1. IF S/G A is to be isolated, THEN perform the following:

   a. Close V1-3A, MSIV.

   b. Verify CLOSED MS-353A, MSIV V1-3A BYP.

   c. Place the "A" FRV controller FCV-478 in MAN AND close FCV-478.

   d. Adjust "A" FRV Bypass Valve controller FCV-479 to close FCV-479.

   e. Close V2-6A, FW HDR SECTION.

   f. Verify CLOSED V2-16A, AFW HDR DISCH Valve.

   g. Verify CLOSED V2-14A, PUMP DISCH Valve.

   h. Verify RV1-1, PORV, setpoint at 1035 psig using UNIT 2 STATUS BOARD pot setting.

   i. Dispatch Operator To The E-1/E-2 To Perform The Following:

      - At MCC-9, verify V2-16A closed AND open breaker V2-16A, MDAFW PUMP HEADER DISCHARGE TO S/G A (CMPT-2ML)

(CONTINUED NEXT PAGE)

CONTINUOUS USE

ATTACHMENT 2

S/G ISOLATION AT HOT SHUTDOWN

(Page 2 of 3)

1. (CONTINUED)

   j. Dispatch Operator To The Aux. Bldg. To Perform The Following:

      - At MCC-10, verify V2-14A closed AND open breaker V2-14A, SDAFW PUMP TO S/G A (CMPT- 3C)
      - At MCC-10, verify V2-16A closed AND open breaker V2-16A, MDAFW PUMP HEADER DISCHARGE TO S/G A (CMPT-4C)
      - At MCC-5, verify V1-8A closed AND open breaker V1-8A, SDAFW PUMP STEAM ISOLATION (CMPT-16F)

2. IF S/G B is to be isolated, THEN perform the following:

   a. Close V1-3B, MSIV.

   b. Close MS-353B, MSIV V1-3B BYP.

   c. Place the "B" FRV controller FCV-488 in MAN AND close FCV-488.

   d. Adjust "B" FRV Bypass Valve controller FCV-489 to close FCV-489.

   e. Close V2-6B, FW HDR SECTION.

   f. Verify CLOSED V2-16B, AFW HDR DISCH Valve.

   g. Verify CLOSED V2-14B, PUMP DISCH Valve.

   h. Verify RV1-2, PORV, setpoint at 1035 psig using UNIT 2 STATUS BOARD pot setting.

   i. Dispatch Operator To E-1/E-2 Room To Perform The Following:

      - At MCC-9, verify V2-14B closed AND open breaker V2-14B, SDAFW PUMP TO S/G B (CMPT- 1C)
      - At MCC-6, verify V1-8B closed AND open breaker V1-8B, SDAFW PUMP STEAM ISOLATION (CMPT-16M)

   j. Dispatch Operator To The Aux. Bldg. To Perform The Following:

      - At MCC-10, verify V2-16B closed AND open breaker V2-16B, MDAFW PUMP HEADER DISCHARGE TO S/G B (CMPT-4F)

CONTINUOUS USE

ATTACHMENT 2

S/G ISOLATION AT HOT SHUTDOWN

(Page 3 of 3)

3. IF S/G C is to be isolated, THEN perform the following:

   a. Close V1-3C, MSIV.

   b. Close MS-353C, MSIV V1-3C BYP.

   c. Place the "C" FRV controller FCV-498 in MAN AND close FCV-498.

   d. Adjust "C" FRV Bypass Valve controller FCV-499 to close FCV-499.

   e. Close V2-6C, FW HDR SECTION.

   f. Verify CLOSED V2-16C, AFW HDR DISCH Valve.

   g. Verify CLOSED V2-14C, PUMP DISCH Valve.

   h. Verify RV1-3, PORV, setpoint at 1035 psig using UNIT 2 STATUS BOARD pot setting.

   i. Dispatch Operator To The E-1/E-2 To Perform The Following:

      - At MCC-9, verify V2-16C closed AND open breaker V2-16C, MDAFW PUMP HEADER DISCHARGE TO S/G C (CMPT-3J)

      - At MCC-6, verify V1-8C closed AND open breaker V1-8C, SDAFW PUMP STEAM ISOLATION (CMPT-18M)

   j. Dispatch Operator To The Aux. Bldg. To Perform The Following:

      - At MCC-10, verify V2-14C closed AND open breaker V2-14C, SDAFW PUMP TO S/G C (CMPT - 4M)

4. Return to Procedure and Step in effect.

- END -

CONTINUOUS USE

ATTACHMENT 3

ESTABLISHING RCS COLD SHUTDOWN BORON CONCENTRATION

(Page 1 of 3)

1. Record the minimum required Mode 5 Boron concentration.

   _______ ppm

2. Determine RCS boron concentration change required to establish CSD boron concentration.

   ___________________ - ___________________ = _______ppm
   Required CSD Boron Concentration — Latest RCS Boron Sample

3. Determine volume of boric acid to be added using Station Curve Book.

   _______ gallons

4. IF the MOV-350, BA TO CHARGING PMP SUCT Valve flowpath is unavailable, THEN Borate the RCS using OP-301, Chemical and Volume Control System (CVCS) Section "RCS Boration Quick Checklist"

   a. WHEN Boration is complete, THEN Go To Step 13.

5. IF the MOV-350, BA TO CHARGING PMP SUCT Valve flowpath will be used, THEN open MOV-350, BA TO CHARGING PMP SUCT Valve.

6. Start Boric Acid Pump aligned for Blended Makeup.

7. Record time boration commenced.

   ____

8. Record flowrate indicated on FI-110, BORIC ACID BYPASS FLOW.

   _______ gpm

9. Determine time required to establish CSD boron concentration.

$$\frac{\text{Boric Acid Volume Required (gal)}}{\text{FI-110 Flowrate (gpm)}} = \_\_\_\_\_\_\_ \text{ minutes}$$

CONTINUOUS USE

ATTACHMENT 4

CONTROLLING SECONDARY CONTAMINATION

(Page 1 of 3)

1. Contact Health Physics personnel to perform the following:

   a. Walkdown secondary systems to catalog sources of secondary leakage for potential waste releases.

   b. Perform radiation surveys in the secondary with emphasis on the Condensate Polishing Demineralizers, and resin columns in the Secondary Sample Room and Condensate Polisher Sample Room.

   c. Establish radiation control boundaries as necessary.

2. Contact Operations Staff to perform the following:

   a. IF NOT already bypassed, THEN determine the level of secondary contamination at which the Condensate Polishing Demineralizers should be bypassed.

   b. Determine the method of removal of contaminated secondary water:

      - Drain the Condenser to the Waste Disposal System using OP-508, Condenser Drain System section "Draining the Condenser to the Waste Disposal System"

      - Release the secondary to the storm drains.

      - Other alternative means such as vendor processing.

   c. IF any secondary radiation monitors are inoperable, THEN elevate repair priority.

   d. IF contaminated, THEN develop a method of removal AND disposal of the resin in the Condensate Polishing Demineralizers.

   e. Evaluate current waste disposal capacity AND determine the need for additional personnel to process waste after plant shutdown.

CONTINUOUS USE

ATTACHMENT 4

CONTROLLING SECONDARY CONTAMINATION

(Page 2 of 3)

3. Contact Chemistry Personnel to perform the following:

   a. Sample secondary systems for activity.

   b. Develop releases, as required, for the secondary systems.

      - Hotwell
      - Feedwater and Condensate Piping
      - Condensate Pump and Condensate Polishing Building Sumps
      - S/G Blowdown

   c. Determine the need to reduce Radiation Monitor Setpoints to provide additional monitoring capability.

   d. Develop R-15 Conversion Factors for determining S/G Leak Rate vs RMS readings.

CONTINUOUS USE

ATTACHMENT 4

CONTROLLING SECONDARY CONTAMINATION

(Page 3 of 3)

NOTE

- ES-33, HEATER 5B EXT STEAM TO PRV-1985, is located approximately 15 feet in the overhead above Heater 4A.
- A valve wrench AND ladder will be required to close ES-33.
- Auxiliary Steam will be lost during the period following isolation of Extraction Steam until the boilers have pressurized.

4. Contact An Operator To Perform The Following:

   a. IF directed by Operations Management, THEN perform the following to bypass the Condensate Polishers:

      1) Place the SECONDARY BYPASS Switch to the OPEN position

      2) Depress the OFF pushbutton for each in service demineralizer

      3) Secure ANY evolution that passes water through the beds, such as a low volume rinse

   b. Stop the Condensate Pit Sump Pump AND the Condensate Polishing Building Sump Pump.

   c. Close ES-33, HEATER 5B EXT STEAM TO PRV-1985, to isolate Main Steam from the Auxiliary Steam System.

   d. Startup the Auxiliary Boilers using OP-401, Auxiliary Heating System.

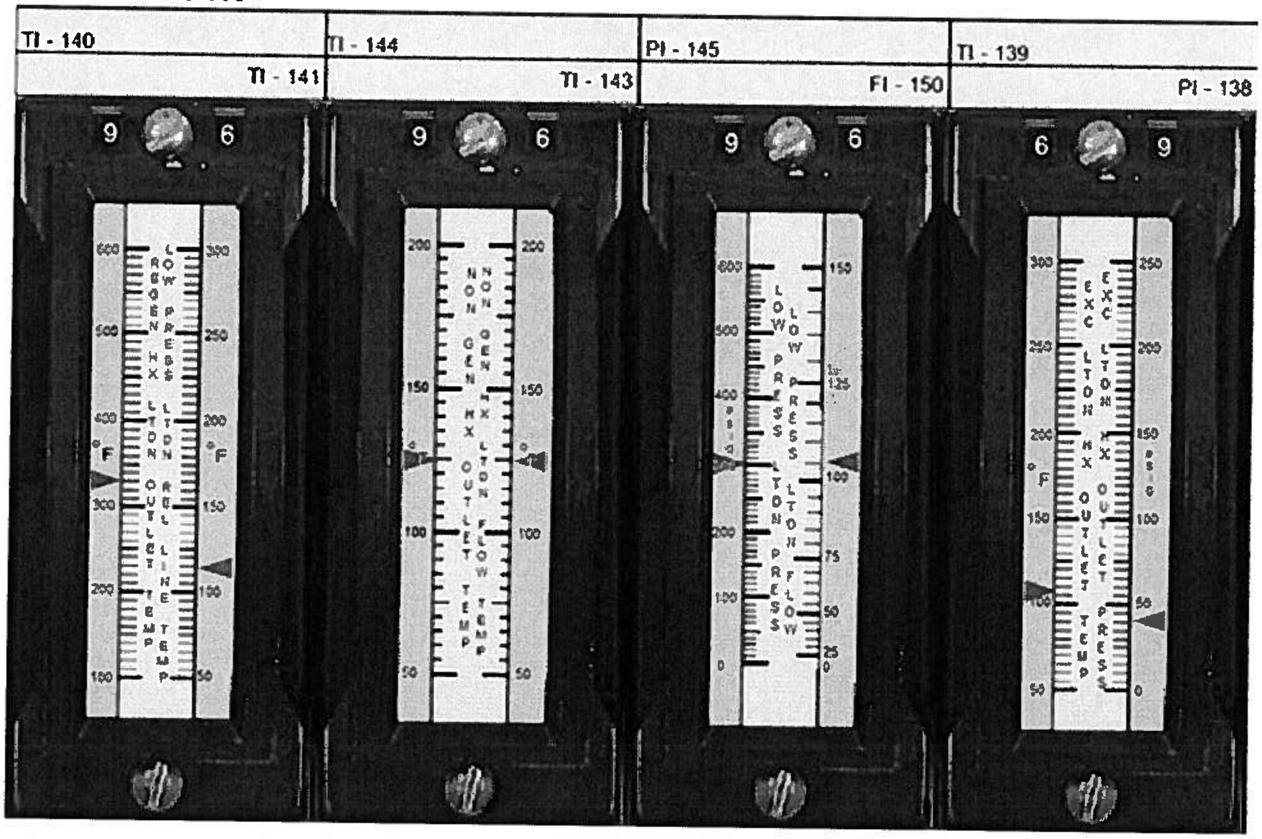
   e. Line up the Auxiliary Steam condensate return to the Auxiliary Boilers as follows:

      - Close AS-327, COND RET TO MISC DRN COLLECTING TK FROM AUX BLDG

      - Open AS-326, COND RET TO COND TK "A" & "B" FROM AUX BLDG

\- END -

84. 076 AA2.05 SRO 001



- Plant is stable at 100% RTP.
- R-9, Letdown Line Area, radiation monitor had a valid alarm and a reading of 15,000 mrem/hr prior to R-9 failing.
- RCS samples indicate an RCS Activity of 550 μCi/gm I-131 Dose Equivalent.

Which ONE(1) of the following completes the statement below?

IAW AOP-005, Radiation Monitoring System, letdown flow __(1)__ and IAW the Emergency Action Level Matrices, an __(2)__ must be declared.
(REFERENCE PROVIDED)

A. (1) may remain as-is

(2) Unusual Event

B. (1) may remain as-is

(2) Alert

C. (1) is required to be reduced

(2) Unusual Event

D✓ (1) is required to be reduced

(2) Alert

The correct answer is D.

A. Incorrect. The given letdown flow reading is indicative of one 60 gpm and one 45 gpm letdown orifices being in service. AOP-005 requires that letdown be reduced to less than or equal to one letdown orifice valve open. Therefore, letdown flow is required to be reduced. The second part is incorrect. The candidate may confuse the 550 μCi/gm I-131 Dose Equivalent reading as being greater than a R-9 reading of 500 mrem/hr on the EAL Matrix for an Unusual Event. The activity reading exceeds the 300 μCi/gm I-131 Dose Equivalent limit for a loss of the Fuel Cladding Barrier which results in meeting the Alert declaration criteria.

B. Incorrect. The given letdown flow reading is indicative of one 60 gpm and one 45 gpm letdown orifices being in service. AOP-005 requires that letdown be reduced to less than or equal to one letdown orifice valve open. Therefore, letdown flow is required to be reduced. The second part of distractor is correct.

C. Incorrect - The given letdown flow reading is indicative of one 60 gpm and one 45 gpm letdown orifices being in service. AOP-005 requires that letdown be reduced to less than or equal to one letdown orifice valve open. Therefore, letdown flow is required to be reduced. The candidate may confuse the 550 μCi/gm I-131 Dose Equivalent reading as being greater than a R-9 reading of 500 mrem/hr on the EAL Matrix for an Unusual Event. The activity reading exceeds the 300 μCi/gm I-131 Dose Equivalent limit for a loss of the Fuel Cladding Barrier which results in meeting the Alert declaration criteria.

D. Correct

Question 84
Tier/Group 1/2
K/A Importance Rating - RO 2.2 SRO 2.5

Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: CVCS letdown flow rate indication

Reference(s) - Sim/Plant design,
Proposed References to be provided to applicants during examination - EAL Matrix
Learning Objective - AOP-005-004, EAL 004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments -

SRO: Candidate must make an EAL Declaration, which is an SRO function.

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. Check Letdown Orifice Isolation Valve(s) - LESS THAN OR EQUAL TO ONE of the following OPEN:

   - CVC-200A
   - CVC-200B
   - CVC-200C

   RESPONSE NOT OBTAINED: Refer to OP-301, CHEMICAL AND VOLUME CONTROL SYSTEM, for direction on removal of an orifice from service.

8. Control Charging Flow To Maintain PZR Level

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

ATTACHMENT 9

AREA MONITOR R-9 - LETDOWN LINE AREA

(Page 2 of 2)

9. Request E&C To Sample The RCS To Determine The Following:

   - Gross Activity
   - Iodine
   - Gaseous Activity

10. Go To The Main Body, Step 1.b, Of This Procedure

- END -

FISSION PRODUCT BARRIER LOSS/POTENTIAL LOSS MATRIX AND TECHNICAL BASES

Table F-1 Fission Product Barrier Matrix

	Fuel Cladding Barrier		Reactor Coolant System Barrier		Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
CSFST	1. CSFST Core Cooling-RED	1. CSFST Core Cooling-ORANGE OR CSFST Heat Sink-RED and heat sink required	None	1. CSFST RCS Integrity - RED OR CSFST Heat Sink - RED and heat sink required	None	1. CSFST Containment - RED
Core Exit T/Cs	2. Core exit T/Cs ≥ 1,200°F	2. Core exit T/Cs ≥ 700°F	None	None	None	2. Core exit T/Cs ≥ 1,200°F AND Core cooling restoration procedures not effective within 15 min. 3. All of the following: - Core exit T/Cs ≥ 700°F - Reactor Vessel water level ≤ Table F-2 thresholds - Core cooling restoration procedures not effective within 15 minutes
Radiation	3. Containment High Range Radiation Monitor R-32A or R-32B > 100 Rem/hr 4. Letdown line area radiation monitor R-9 > 25,000 mRem/hr	None	1. Containment High Range Radiation Monitor R-32A or R-32B > 5 Rem/hr	None	None	4. Containment High Range Radiation Monitor R-32A or R-32B > 2000 Rem/hr
Inventory	None	3. Reactor Vessel water level ≤ Table F-2 thresholds	2. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling less than 35°F [55°F] 3. SGTR that results in an ECCS (SI) actuation	2. Unisolable RCS leak exceeding the capacity of one charging pump (77 gpm)	1. Rapid unexplained Containment pressure drop following initial increase 2. Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions 3. Ruptured S/G is also faulted outside of Containment 4. Primary-to-secondary leakage > 10 gpm with non-isolable steam release from affected S/G to the environment	5. Containment pressure 42 psig and increasing 6. Containment hydrogen concentration ≥ 4% 7. Containment pressure ≥ 10 psig with < one full train of depressurization equipment operating Note: One Containment Spray System train and one Containment Cooling System train comprise one full train of depressurization equipment
Other	5. Coolant activity > 300 μCi/gm I-131 Dose Equivalent	None	None	None	5. Containment isolation valve(s) not closed after Containment isolation AND Downstream pathway to the environment exists	None
Judgment	6. Any condition in the opinion of the SEC that indicates loss of the Fuel Clad barrier	4. Any condition in the opinion of the SEC that indicates potential loss of the Fuel Clad barrier	4. Any condition in the opinion of the SEC that indicates loss of the RCS barrier	3. Any condition in the opinion of the SEC that indicates potential loss of the RCS barrier	6. Any condition in the opinion of the SEC that indicates loss of the Containment barrier	8. Any condition in the opinion of the SEC that indicates potential loss of the Containment barrier

The specified value of 100 Rem/hr is conservatively at the low end of the calculated range. This value is higher than that specified for RCS barrier Loss #3.

It is important to recognize that the radiation monitor may be sensitive to shine from the Reactor Vessel or RCS piping.

Monitors used for this Fission Product Barrier Loss threshold are the Containment High Range Radiation Monitors R-32A and R-32B. These monitors provide indication in the Control Room with a range of 1E0 to 1E7 Rem/hr (Ref. 3).

RNP References:

1. EPTSC-07, Damage Assessment
2. RNP-M/MECH-1744, R-32A and R-32B Calculation for Core Damage Assessment
3. UFSAR Section 12.3.3.1.2.2
4. OMM-014, Radiation Monitor Setpoints

4. **Letdown line area radiation monitor R-9 > 25,000 mRem/hr**

The normal CVCS charging and letdown flowpath allows purification of the reactor coolant and control of the RCS volume. Hot (547°F) reactor coolant from the cold leg of loop 1 passes through the regenerative heat exchanger. The discharge of the regenerative heat exchanger then passes through the non-regenerative heat exchanger. Downstream of the non-regenerative heat exchanger and upstream of the mixed bed demineralizers, the letdown stream passes by area radiation monitor R-9, which is mounted above the letdown line pipe. In order for R-9 readings to represent fission product activity in the reactor coolant and thereby warn of potential fuel element failure, letdown must be in service allowing flow through the letdown line and past the radiation monitor.

Fuel failure in excess of 5% or 300 μCi/gm I-131 Dose Equivalent will trigger the threshold value of 25,000 mR/hr (Ref.1).

RNP References:

1. RNP-M-MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels
2. OMM-014, Radiation Monitor Setpoints

5. **Coolant activity > 300 μCi/gm I-131 Dose Equivalent**

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold Dose Equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 5% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost.

RNP References:

1. RNP-M-MECH-1745, Calculation of Setpoints for Accident Rad Monitors and EP Declaration Levels

6. **Any condition in the opinion of the SEC that indicates Loss of the Fuel Clad barrier**

The Site Emergency Coordinator (SEC) judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all Fission Product Barriers and likely entry to the EOPs. The SEC should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

RNP References:

None

DISCUSSION (From the WOG FR-Z.2 Basis Document)

1. INTRODUCTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, is a Function Restoration Guideline (FRG) that provides procedural guidance when the containment level is greater than flood level.

There is only one explicit transition to guideline FR-Z.2. It is from the Critical Safety Function Status Tree F-0.5, CONTAINMENT, on an ORANGE priority when containment sump level is greater than flood level.

After all the actions in guideline FR-Z.2 are completed, the operator is instructed to return to the guideline and step in effect.

2. DESCRIPTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, provides actions to respond when the containment level is greater than design flood level. This level is significant since the critical systems and components, which are necessary to ensure an orderly safe plant shutdown and provide feedback to the operator regarding plant conditions, are normally located above the design flood level. Therefore, the guideline FR-Z.2 is entered from the Containment Status Tree on an ORANGE priority when this design flood level is exceeded.

The primary purpose of the containment sump area is to collect the water injected into the containment or spilled from the reactor coolant system following an accident. The water collected in the containment sump is then available for long term core and/or containment cooling via the emergency core cooling or containment spray recirculation systems. In addition, the containment sump collects the injected or spilled water into areas such that vital systems or components will not be flooded and thus rendered inoperable.

The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators. This water volume approximates the maximum water volume introduced into the containment following a LOCA plus a steamline or feedline break inside containment.

An indicated water level in the containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes have been introduced into the containment. Also, the high water level provides an indication that potential flooding of critical systems and components needed for plant recovery may occur.

The actions in this guideline attempt to identify any unexpected source of water and isolate it if possible. Beyond that the plant engineering staff is consulted to determine if transfer of containment sump water to other tanks is appropriate.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-Z.2 is to provide actions to respond to containment flooding.

The following subsections provide a summary of the major action categories of operator actions and the key utility decision points for guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-Z.2 is given below in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN FR-Z.2

- o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible
- o Notify Plant Engineering Staff of Sump Level and Activity Level

o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible

The first action in this guideline is to try to identify the source of water which is causing containment flooding and isolate it. The concern regarding flooding is that critical plant components needed for plant recovery could be damaged and rendered inoperable.

o Notify Plant Engineering Staff of Sump Level and Activity Level

By knowing the sump level and activity level, the plant engineering staff can determine if the excess water can be transferred to storage tanks located outside containment.

85. W/E 09 EA2.1 SRO 001
Given the following plant conditions:

- A loss of off-site power and reactor trip has occurred.
- The crew performed actions of PATH-1 and has transitioned to EPP-4, Reactor Trip Response.
- A rupture in the Instrument Air System at time of trip has resulted in Instrument Air pressure lowering to 20 psig.
- CETC temperature is 555°F and rising 15 minutes after the trip.
- S/G pressure is 1075 psig and rising.

Which ONE (1) of the following completes the statements below?

RCS Natural Circulation __(1)__ exist(s).
In the situation described above, **detailed** steps regarding the control of RCS temperature will be provided in __(2)__.

A. (1) does

(2) AOP-017, Loss of Instrument Air

B. (1) does

(2) EPP-4, Reactor Trip Response

C. (1) does NOT

(2) EPP-4, Reactor Trip Response

D✓ (1) does NOT

(2) AOP-017, Loss of Instrument Air

Correct answer is D.

A. Incorrect. NC does not exist with Tave at 555°F and S/G pressures at 1185 psig. Also, nitrogen must be manually aligned to S/G PORVs on a loss of air.

B. Incorrect. NC does not exist. AOP-017 provides the procedural direction on how to control S/G pressure and RCS temperature on a loss of instrument air. AOP-017 is a concurrent use procedure.

C. Incorrect. Nitrogen must be manually aligned to S/G PORVs on a loss of Air in accordance with AOP-017.

D. Correct.

Question 85
Tier/Group 1/2
K/A Importance Rating - RO 3.1 SRO 3.8

Ability to determine and interpret the following as they apply to the (Natural Circulation Operations): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Reference(s) - Sim/Plant design, Supplement E, AOP-017, EPP-4, EPP-5
Proposed References to be provided to applicants during examination - None
Learning Objective - EPP-5-004
Question Source - BANK (Last used on 2007 NRC Exam. Format of question modified significantly.)
Question Cognitive Level - H
10 CFR Part 55 Content - 43.5 / 45.13
Comments -

SRO: Candidate must have knowledge of when to implement attachments within abnormal / emergency procedures. Candidate must know that the steps to control RCS temperature using nitrogen aligned to the S/G PORVs is contained in AOP-017 and not EPP-4. Procedure selection and knowledge of procedure content versus knowledge of procedure mitigation strategy.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

Supplement E

Natural Circulation Verification

(Page 1 of 1)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Natural Circulation Status As Follows: • Steam pressure - STABLE OR DECREASING AND • RCS subcooling - GREATER THAN 35°F [55°F] AND • Core exit T/Cs - STABLE OR DECREASING AND • RCS hot leg temperatures - STABLE OR DECREASING AND • RCS cold leg temperatures - TRENDING TO SATURATION TEMPERATURE FOR STEAM PRESSURE	Raise the rate of dumping steam.
2.	Return To Procedure And Step In Effect	

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

SECTION B

HOT SHUTDOWN (WITHOUT RHR IN SERVICE)

(Page 1 of 10)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
* 1.	Determine If IA Capacity Has Been Restored As Follows:	
	a. Check IA Header pressure: • GREATER THAN 85 PSIG AND • STABLE OR RISING	a. IF IA capacity is restored, THEN Go To Step 1.b. Go To Step 2.
	b. Go To Attachment 4, Restoration From Loss of Instrument Air	
2.	Control RCS Temperature By Dumping Steam Using One Of The Following Methods Listed In Order Of Preference: a. Steam Dump to Condenser b. Steam Line PORVs controlled by IA c. Steam Line PORVs controlled by nitrogen using Attachment 2, Nitrogen Alignment to Steam Line PORVs d. Manually steam each intact S/G using Attachment 3, Manual Steam Dump of S/Gs	
* 3.	Check S/G Level Control - ADVERSELY AFFECTED BY LOSS OF IA	IF any S/G level is affected, THEN perform Step 4. Observe NOTE prior to Step 5 and Go To Step 5.
4.	Establish AFW Flow To Maintain S/G Levels At Desired Level	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

NITROGEN ALIGNMENT TO STEAM LINE PORVs

(Page 1 of 5)

NOTE

A Locked Valve Key is required to perform this attachment.

* 1. Check MSIVs As Follows:

a. Check MSIVs - CLOSED

a. IF the MSIVs close, THEN Perform step 1.b.

Go To Step 2.

b. Place the MSIV Control Switches in the CLOSE position

2. Check Steam Line PORVs - TO BE CONTROLLED FROM RTGB

Go To Step 9.

NOTE

- Steps 3 through 7 align backup nitrogen to control Steam Line PORVs and ensure the MSIVs remain closed.
- IA-297 is located at the Southeast corner of Pipe Jungle at an elevation between the feed and steam lines.

3. Close IA-297, HDR STOP TO PORV STATION & MSIV'S

NOTE

SDN-13 is located at the Steam Dump Nitrogen Accumulator.

4. Open SDN-13, NITROGEN BACKUP SUPPLY

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

NITROGEN ALIGNMENT TO STEAM LINE PORVs

(Page 2 of 5)

> **NOTE**
>
> SDN-28 and SDN-29 are located at the Southeast corner of Pipe Jungle on the Mezzanine Deck.

5. Unlock AND Close SDN-29, STEAM LINE PORV NITROGEN B/U TO IA TELL-TALE DRAIN

6. Unlock AND Open SDN-28, NITROGEN BACKUP TO STEAM LINE PORVS

> **NOTE**
>
> IA-423 is located at the Southeast corner of Pipe Jungle on the Mezzanine Deck.

7. Unlock AND Open IA-423, NITROGEN TO STEAM LINE PORVS

8. Go To Section And Step In Effect

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

NITROGEN ALIGNMENT TO STEAM LINE PORVs

(Page 3 of 5)

9. Transfer Steam Line PORV Control To The Local Controllers At The Secondary Control Panel As Follows:

   a. Place PIC-477, manual thumbwheel, to the closed position by rotating the white thumbwheel in the up direction

   b. Place PORV RV-1 Switch in the DEFEAT position

   c. Place the transfer switch on PIC-477 to MAN position

   d. Place PIC-487, manual thumbwheel, to the closed position by rotating the white thumbwheel in the up direction

   e. Place PORV RV-2 Switch in the DEFEAT position

   f. Place the transfer switch on PIC-487 to MAN position

   g. Place PIC-497, manual thumbwheel, to the closed position by rotating the white thumbwheel in the up direction

   h. Place PORV RV-3 Switch in the DEFEAT position

   i. Place the transfer switch on PIC-497 to MAN position

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

NITROGEN ALIGNMENT TO STEAM LINE PORVs

(Page 4 of 5)

NOTE

- Steps 10 through 14 align backup nitrogen to control Steam Line PORVs and ensure the MSIVs remain closed.
- IA-297 is located at the Southeast corner of Pipe Jungle at an elevation between the feed and steam lines.

10. Close IA-297, HDR STOP TO PORV STATION & MSIV'S

NOTE

SDN-13 is located at the Steam Dump Nitrogen Accumulator.

11. Open SDN-13, NITROGEN BACKUP SUPPLY

NOTE

SDN-28 and SDN-29 are located at the Southeast corner of Pipe Jungle on the Mezzanine Deck.

12. Unlock AND Close SDN-29, STEAM LINE PORV NITROGEN B/U TO IA TELL-TALE DRAIN

13. Unlock AND Open SDN-28, NITROGEN BACKUP TO STEAM LINE PORVS

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 2

NITROGEN ALIGNMENT TO STEAM LINE PORVs

(Page 5 of 5)

NOTE

IA-423 is located at the Southeast corner of Pipe Jungle on the Mezzanine Deck.

14. Unlock AND Open IA-423, NITROGEN TO STEAM LINE PORVS

15. Control RCS Temperature From The Secondary Control Panel By Opening And Closing The PORVs As Follows:

   - To open a PORV, rotate the white manual thumbwheel of the desired pressure indicating controller in the DOWN direction

   - To close a PORV, rotate the white manual thumbwheel of the desired pressure indicating controller in the UP direction

16. Go To Section And Step In Effect

\- END -

CONTINUOUS USE

ATTACHMENT 3

MANUAL STEAM DUMP OF S/Gs

(Page 1 of 4)

NOTE

- OP-923, Containment Integrity, provides the controls for opening valves that serve as Containment Integrity.
- A Locked Valve Key is required to perform this attachment.

1. Align the following drains located at the Pipe Jungle:
   - Unlock and open MS-19, SG "A" STEAM LINE BEFORE SEAT DRN ROOT ISOL
   - Open MS-19A, SG "A" STEAM LINE BEFORE SEAT DRN ISOL
   - Throttle MS-40, SG "A" STEAM LINE BEFORE SEAT DRAIN ISOL
   - Open MS-21, SG "A" STEAM STOP V1-3A AFTER SEAT DRN ROOT ISOL
   - Throttle MS-43, SG "A" STEAM STOP V1-3A AFTER SEAT DRAIN ISOL
   - Unlock and open MS-28, SG "B" STM LINE BEFORE SEAT DRAIN ROOT ISOL
   - Open MS-28A, SG "B" STM LINE BEFORE SEAT DRAIN ISOL
   - Throttle MS-41, SG "B" STEAM LINE BEFORE SEAT DRAIN ISOL
   - Open MS-30, SG "B" STEAM STOP V1-3B AFTER SEAT DRAIN ROOT ISOL
   - Throttle MS-44, SG "B" STEAM STOP V1-3B AFTER SEAT DRAIN ISOL
   - Unlock and open MS-37, SG "C" STEAM LINE BEFORE SEAT DRAIN ROOT ISOL
   - Open MS-37A, SG "C" STEAM LINE BEFORE SEAT DRAIN ISOL
   - Throttle MS-42, SG "C" STEAM LINE BEFORE SEAT DRAIN ISOL
   - Open MS-39, SG "C" STEAM STOP V1-3C AFTER SEAT DRAIN ROOT ISOL

(CONTINUED NEXT PAGE)

CONTINUOUS USE

ATTACHMENT 3

MANUAL STEAM DUMP OF S/Gs

(Page 2 of 4)

1. (CONTINUED)

   - Throttle MS-45, SG "C" STEAM STOP V1-3C AFTER SEAT DRAIN ISOL

CONTINUOUS USE

ATTACHMENT 3

MANUAL STEAM DUMP OF S/Gs

(Page 3 of 4)

2. IF additional steaming is required, THEN perform the following, as necessary, to control RCS temperature:

   a. Unlock and close the breakers for the MSIV BYP Valves on MCC-8:

      - MS-353A, MSIV V1-3A BYP (CMPT 1C)
      - MS-353B, MSIV V1-3B BYP (CMPT 2C)
      - MS-353C, MSIV V1-3C BYP (CMPT 3C)

   b. Open the following MSIV BYPs from the RTGB:

      - MS-353A, MSIV V1-3A BYP
      - MS-353B, MSIV V1-3B BYP
      - MS-353C, MSIV V1-3C BYP

   c. Throttle open the Seventy Two Inch Header Vent Valves:

      - MS-231, 72" HEADER ATMOS VENT
      - MS-232, 72" HEADER ATMOS VENT

   d. Perform the following on the North Header (located on the mezzanine level, above MCC-8) (ladder required):

      1) Open the North Header Drain and Vent Valves:

         - MS-98, NORTH MAIN STEAM LINE DRAIN ROOT ISOLATION
         - MS-99, NORTH MAIN STEAM LINE VENT ISOL

      2) Throttle MS-100, NORTH MAIN STEAM LINE VENT TO ATMOS, as directed by the RTGB operator.

(CONTINUED NEXT PAGE)

CONTINUOUS USE

ATTACHMENT 3

MANUAL STEAM DUMP OF S/Gs

(Page 4 of 4)

2. (CONTINUED)

   e. Perform the following on the South Header (located on the mezzanine level, above and between MCC 14 and Lube Oil Vapor Extractor) (ladder required):

      1) Open the South Header Drain and Vent Valves:

         - MS-94, SOUTH MAIN STEAM LINE DRN ROOT ISOL
         - MS-95, SOUTH MAIN STEAM LINE VENT ISOL

      2) Throttle MS-96, SOUTH MAIN STEAM LINE VENT TO ATMOS ISOLATION, as directed by the RTGB operator.

- END -

86. 003 A2.03 SRO 001

Given the following plant conditions:

Initial Conditions:
- Plant is operating at 100% RTP.
- "A" RCP Motor Upper Thrust Bearing temperature indicates 230°F.
- "C" RCP Motor Upper Guide Bearing temperature indicates 205°F.

Current Conditions:
- RTBs open.
- S/G Levels within normal band.
- Pressurizer pressure is normal.

Which ONE (1) of the following correctly describes whether RCPs "A" and/or "C" are required to be shutdown and whether the conditions of LCO 3.4.5, RCS Loops - Mode 3, are met?

A. Both "A" and "C" RCPs are required to be secured.

Conditions of LCO 3.4.5 met.

B. "A" RCP is required to be secured. "C" RCP is NOT required to be secured.

Conditions of LCO 3.4.5 met.

C.✓ Both "A" and "C" RCPs are required to be secured.

Conditions of LCO 3.4.5 NOT met.

D. "A" RCP is required to be secured. "C" RCP is NOT required to be secured.

Conditions of LCO 3.4.5 NOT met.

The correct answer is C.

A. Incorrect. The first part of distractor is correct. Both AOP-018 and AOP-014 state that the RCP is to be tripped if any RCP motor bearing temperature is greater than 200°F. The conditions of LCO 3.4.5 are NOT met. LCO 3.4.5 requires that two RCS loops be operable and one RCS loop shall be in operation as long as the RTBs are open. Candidate may think that the loop is operable since the S/Gs are greater than 16%. However, LCO 3.4.5 bases states that an Operable RCS loop consists of one Operable RCP and one Operable S/G, which has a minimum water level specified in SR 3.4.5.2 (>= 16%). An RCP is Operable if it is capable of being powered and is able to provide forced flow if required. Since the B and C RCPs had to be secured due to high motor bearing temperatures then they are not available for operation.

B. Incorrect. Candidate incorrectly assumed that operation of "C" RCP is allowed at greater than 200°F Motor Bearing Temperature. The plausibility for this is that the limit for RCP Pump Bearing temperature is 225°F. See "A" above for remaining justification.

C. Correct.

D. Incorrect. Candidate incorrectly assumed that operation of "C" RCP is allowed at greater than 200°F Motor Bearing Temperature. The plausibility for this is that the limit for RCP Pump Bearing temperature is 225°F. It is plausible for the candidate to apply the conditions of LCO 3.4.4, RCS Loops - Modes 1 and 2, to this situation and think that 3.4.5 are NOT met. LCO 3.4.4 requires that all three loops be operable and in operation. Also, the candidate could have simply assumed that LCO 3.4.5 required all three loops to be Operable.

Question 86
Tier/Group 2/1
K/A Importance Rating - RO 2.7 SRO 3.1

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

Reference(s) - Sim/Plant design, APP-001, AOP-014, AOP-018, ITS 3.4.4, 3.4.5
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-014-004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments -

SRO: Motor parameter being out of spec (above manual trip setpoint) requires RCPs to be secured, which renders RCS loops inoperable. Procedures, in the form of Tech Specs and AOPs, must be applied to mitigate the conditions. The question would qualify as SRO-only because the applicant must apply basis information to determine whether the loop is operable.

Reviewed and approved by MAB.

87. 007 A2.05 SRO 001

Given the following plant conditions:

- Plant is operating at 90% RTP following a refueling outage.
- PCV-455C, PZR PORV, has developed **excessive** seat leakage.
- APP-003-B3, PRT HI TEMP AND APP-003-C3, PRT HI PRESS, has alarmed.
- PRT Pressure is 6.1 psig and rising.
- RC-536, PZR PORV BLOCK, has been closed to isolate the leakage.

Which ONE (1) of the following completes the statements below?

IAW ITS LCO 3.4.11 Bases, PCV-455C __(1)__ operable and power __(2)__ required to be removed from RC-536.

A.✓ (1) is NOT

(2) is NOT

B. (1) is

(2) is

C. (1) is

(2) is NOT

D. (1) is NOT

(2) is

The correct answer is A.

A. Correct.

B. Incorrect. ITS LCO 3.4.11 bases states that an Operable PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. This is a recent change to the ITS bases. In the past the bases did not address seat leakage. The PORV was only required to be capable of manually opening and closing to be considered operable. Candidate may think that since the PORV is leaking and must be isolated that it would be prudent to remove power to prevent inadvertent reinitiation of the PORV seat leakage.

C. Incorrect. ITS LCO 3.4.11 bases states that an Operable PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. This is a recent change to the ITS bases. In the past the bases did not address seat leakage. The PORV was only required to be capable of manually opening and closing to be considered operable. For this distractor the candidate incorrectly determines that the PORV is Operable. Therefore, it would seem logical that the PORV block valve would not have power removed.

D. Incorrect. The first part of the distractor is correct. The second part of the distractor would be correct if the PORV was inoperable and not capable of being manually cycled. In the situation given, the PORV is inoperable solely on the fact that it has excessive seat leakage.

Question 87
Tier/Group 2/1
K/A Importance Rating - RO 3.2 SRO 3.6

Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Exceeding PRT high-pressure limits

Reference(s) - Sim/Plant design, AOP-019, AOP-016
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-016-002, AOP-019-002, PZR004
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A match because candidate must determine the correct actions to take for a condition given that could lead to exceeding the pressure limits of the PRT.

SRO: The candidate must demonstrate knowledge of TS bases to determine the operability of a plant component.

Reviewed and approved by MAB.

88. 022 G2.4.50 SRO 001

Given the following plant conditions:

- The plant is operating at 100% RTP.
- HVH-1 is OOS due to bearing damage.
- "B" CV Spray Pump is OOS due to a motor calculation error.
- APP-002-A7, HVH-1/2/3/4 HI VIB, is received and the HIGH VIBRATION white light is illuminated for HVH-3.

Which ONE (1) of the following completes the statements below?

To verify the alarm the operator should __(1)__.

If the alarm is valid, the most limiting ITS LCO condition(s) met requires that __(2)__.

(REFERENCE PROVIDED)

A. (1) stop HVH-3, depress the vibration switch reset pushbutton, and then restart HVH-3 RO

(2) the containment spray train AND one containment cooling train be restored to OPERABLE status in 72 hours

B.✓ (1) depress the vibration switch reset pushbutton

(2) entry into LCO 3.0.3 be performed immediately

C. (1) stop HVH-3, depress the vibration switch reset pushbutton, and then restart HVH-3

(2) entry into LCO 3.0.3 be performed immediately

D. (1) depress the vibration switch reset pushbutton

(2) the containment spray train AND one containment cooling train be restored to OPERABLE status in 72 hours

The correct answer is B.

A. Incorrect. The fan does not need to be stopped to reset the vibration alarm. Plausible since some components are required to be stopped to reset various functions. The second half of the distractor is correct.

B. Correct.

C. Incorrect. The fan does not need to be stopped to reset the vibration alarm. Plausible since some components are required to be stopped to reset various functions. The second half of the distractor is also incorrect because HVH-1 and HVH-2 makeup one train of containment cooling. Candidate may think that two HVH units being OOS equates to two containment trains being OOS.

D. Incorrect. The first half of the distractor is correct. The second half of the distractor is incorrect because HVH-1 and HVH-2 makeup one train of containment cooling. Candidate may think that two HVH units being OOS equates to two containment trains being OOS.

Question 88
Tier/Group 2/1
K/A Importance Rating - RO 4.2 SRO 4.0

Containment Cooling System (CCS): Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Reference(s) - Sim/Plant design, APP-002-A7, ITS 3.6.6
Proposed References to be provided to applicants during examination - ITS 3.6.6
Learning Objective - CVHVAC 008
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.3
Comments -

SRO: Application of Required Actions of Tech Specs.

ALARM

HVH-1/2/3/4 HI VIB

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Fan Starting
2. Fan out of balance
3. Fan loose at pedestal
4. Vibration Switch failure

OBSERVATIONS

1. High Vibration alarm lights on RTGB.
2. HVH-1, 2, 3, & 4 AIR FLOW LOST Annunciators (APP-002-A5, B5, C5, & D5)

ACTIONS

CK (✓)

1. **IF** the AIR FLOW LOST annunciator is also illuminated, **THEN STOP** the affected fan. ____
2. **ATTEMPT** to Reset High Vibration alarm. ____
   a) **IF** the alarm will **NOT** reset **AND** the fan is required for plant operation, **THEN DISPATCH** personnel to check the affected fan vibration. ____
   b) **IF** the fan is **NOT** required for plant operation, **THEN STOP** the affected fan. ____
4. **IF** available, **THEN START** a standby Containment Recirc Cooler Fan. ____
5. **IF** local checks confirm high vibration, **THEN CONTACT** Engineering for investigation. ____

DEVICE/SETPOINTS

1. Vibration Switch /

POSSIBLE PLANT EFFECTS

1. Loss of HVH Fan
2. Possible entry into TECH SPEC LCO
3. CV elevated temperature.

REFERENCES

1. ITS LCO 3.6.6
2. CWD B-190628, Sh 511R

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours DISTRACTOR AND 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. Two containment spray trains inoperable. OR Any combination of three or more trains inoperable.	F.1 Enter LCO 3.0.3.	Immediately

Distractor

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Operate each containment cooling train fan unit for ≥ 15 minutes.	31 days
SR 3.6.6.3	Verify cooling water flow rate to each cooling unit is ≥ 750 gpm.	31 days
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.7	Verify each containment cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.8	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage

89. 059 A2.03 SRO 001

Given the following plant conditions:

- The plant has experienced a Small Break LOCA inside Containment.
- PATH-1 is currently being implemented.
- The BOP became distracted and allowed AFW to feed the S/Gs to the following Narrow Range levels.
  - "A" S/G: 2%
  - "B" S/G: 10%
  - "C" S/G: 86%

Which ONE (1) of the following completes the statement below?

The major concern with "C" S/G is the __(1)__ and isolation of feedwater to "C" S/G is required to be performed IAW __(2)__.

A.✓ (1) increased dead weight and water hammer effects on main steamlines

(2) FRP-H.3, Response to Steam Generator High Level

B. (1) increased dead weight and water hammer effects on main steamlines

(2) Supplement G, Steam Generator Isolation

C. (1) increased dead weight on the S/G vessel external supports in the CV

(2) FRP-H.3, Response to Steam Generator High Level

D. (1) increased dead weight on the S/G vessel external supports in the CV

(2) Supplement G, Steam Generator Isolation

The correct answer is A.

A. Correct.

B. Incorrect. The first part of the distractor is correct. The second half of the distractor would reduce S/G level, however, FRP-H.3 has the operator isolate steam prior to draining the S/G.

C. Incorrect. The S/G supports would experience a higher load, however, the S/G supports are designed to handle full S/Gs as experienced during cold shutdown conditions when S/Gs are placed in wet layup. The steam line supports may be challenged and the need for additional bracing is addressed in FRP-H.3. The second half of the distractor is correct.

D. Incorrect. The S/G supports would experience a higher load, however, the S/G supports are designed to handle full S/Gs as experienced during cold shutdown conditions when S/Gs are placed in wet layup. The steam line supports may be challenged and the need for additional bracing is addressed in FRP-H.3. The second half of the distractor would reduce S/G level, however, FRP-H.3 has the operator isolate steam prior to draining the S/G.

The S/G supports would experience a higher load, however, the S/G supports are designed to handle full S/Gs as experienced during cold shutdown conditions when S/Gs are placed in wet layup. The steam line supports may be challenged and the need for additional bracing is addressed in FRP-H.3.

Question 89
Tier/Group 1/1
K/A Importance Rating - RO 3.7 SRO 3.9

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

Reference(s) - Sim/Plant design, APP-006-F2, FRP-H.3, FRP-H.3BD, Supplement G.
Proposed References to be provided to applicants during examination - None
Learning Objective - FRP-H.3-003
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - Originally had "increased dead weight on the steam generator vessel supports." However, every validator chose this distractor due to it being so similar to the concern with the structural concerns associated with the steam lines.

SRO: Assessing plant conditions and then selecting a procedure to mitigate, recover, or with which to proceed. Procedure selection that is not a major EOP or Red/Orange FRP. The procedure required to be selected is a Yellow priority FRP. Also, testing the knowledge of the basis for the FRP.

ALARM

S/G C NAR RANGE HI LEVEL *** WILL REFLASH ***

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Instrument Channel Failure:
   1) Steam Flow
   2) Feedwater Flow
   3) Steam Generator Level
2. Level Control System Failure
3. Increase in Steam Flow
4. Excessive Auxiliary Feedwater Flow

OBSERVATIONS

1. Steam Generator Level
2. Feedwater Flow
3. Steam Flow
4. Turbine First Stage Pressure
5. AFW Pump Discharge Flows

ACTIONS

		CK (✓)
1.	**IF** an Instrument Channel supplying SGLC has failed, **THEN** refer to AOP-025.	______
2.	**IF** Feedwater Control failure, **THEN** refer to AOP-010.	______
3.	**IF** an Instrument Channel **NOT** supplying SGLC has failed, **THEN** removed from service using OWP-027.	______

DEVICE/SETPOINTS

1. LC-494, LC-495, LC-496 / 60% of span
2. LC-494-1(X-1), LC-494-1(X-2), LC-495-1(X-1), LC-495-1(X-2), LC-496-1(X-1), LC-496-1(X-2) / 75% of span (reflash)

POSSIBLE PLANT EFFECTS

1. FW Isolation signal occurs at 75% causing a Turbine Trip.

REFERENCES

1. ITS Table 3.3.1-1 Item 13, Table 3.3.3-1 Item 13
2. AOP-010, Main Feedwater/Condensate Malfunction
3. AOP-025, RTGB Instrument Failure
4. OWP-027, Steam Generator Level
5. CWD B-190628: Sheet 399, Cable L; Sheet 417, Cable M;
   Sheet 420, Cables K, Q

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 4

FUNCTION RESTORATION PROCEDURE

FRP-H.3

RESPONSE TO STEAM GENERATOR HIGH LEVEL

REVISION 10

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

   This procedure provides actions to respond to a steam generator high level condition and to address the steam generator overfill concern.

2. ENTRY CONDITIONS

   a. CSF-3, Heat Sink Critical Safety Function Status Tree on a YELLOW condition.

   b. FRP-H.2, Response To Steam Generator Overpressure, if the affected S/G level is high.

   c. FRP-H.4, Response To Loss Of Normal Steam Release Capability, if the affected S/G level is high.

- END -

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

Steam release from any S/G with level greater than 84% [82%] could result in damage to the Steamlines.

****

NOTE

Throughout this procedure, "affected" refers to any S/G in which level is greater than 75%. "Overfilled" refers to any S/G in which level is greater than 84% [82%].

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Any S/G level - GREATER THAN 75%	Reset SPDS AND return to procedure and step in effect.
* 2.	Determine If A S/G Evaluation Is Needed:	
	a. Check Any S/G - GREATER THAN 84% [82%]	a. IF level increases above 84% [82%], THEN perform Step 2.b. Go To Step 3.
	b. Contact Operations Staff to have an evaluation performed on the S/G for overfilled considerations	
3.	Establish FW Isolation As Follows:	
	a. Verify FW PUMPS A AND B - STOPPED	
	b. Verify affected S/G(s) FW REG Valves - CLOSED	
	c. Verify affected S/G(s) FW REG BYP Valves - CLOSED	
	d. Verify affected S/G(s) FW HDR SECTION Valves- CLOSED	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
4.	Isolate AFW Flow To Affected S/G(s)	
5.	Check Affected S/G(s) Level:	
	a. S/G level - LESS THAN 84% [82%]	a. Go To Step 6.
	b. S/G level - DECREASING	b. Go To Step 6.
	c. Control AFW flow to maintain level between 8% [18%] and 50%	
	d. Reset SPDS <u>AND</u> return to procedure and step in effect	
6.	Isolate Overfilled S/G(s) As Follows:	
	a. Verify affected S/G(s) PORV controller setpoint to 1035 psig	
	b. Verify the affected S/G(s) MSIV <u>AND</u> MSIV BYP - CLOSED	
	c. Maintain steam supply to the SDAFW Pump from at least one S/G	
	d. Verify the affected S/G(s) STEAM SHUTOFF to SDAFW Pump - CLOSED	
	e. Locally verify the affected S/G(s) Warmup Steam Supply Valve to SDAFW Pump - CLOSED	
	f. Locally verify the affected S/G(s) MSIV Above and Below Seat Drains - CLOSED	
7.	Contact Operations Support Staff To Evaluate Need To Brace Steam Lines Due To Dead Weight Loads Caused By Water In The Lines, While Continuing With This Procedure	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

NOTE

The POST ACCIDENT SAMPLING PHASE A CV ISOLATION OVERRIDE Key Switches may be used to monitor affected S/G(s) blowdown for radiation.

8.	Check The Following Radiation Monitors For The Affected S/G(s) -INCREASING OR IN ALARM • R-19(s), S/G Blowdown Radiation Monitor OR • R-31(s), S/G Steamline Radiation Monitor	Go To Step 11.
9.	Check Any Of The Following SGTR EPPs - IN EFFECT: • EPP-12, Post-SGTR Cooldown Using Backfill • EPP-13, Post-SGTR Cooldown Using Blowdown • EPP-14, Post-SGTR Cooldown Using Steam Dump • EPP-17, SGTR With Loss Of Reactor Coolant: Subcooled Recovery • EPP-18, SGTR With Loss Of Reactor Coolant: Saturated Recovery • EPP-19, SGTR Without Pressurizer Pressure Control	Reset SPDS AND Go To PATH-2, Entry Point J.
10.	Reset SPDS AND Return To Procedure And Step In Effect	

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

****

CAUTION

During performance of subsequent steps to drain the affected S/G(s), unisolation of the steam release paths from overfilled S/G(s) before the overfill evaluation has been completed could result in damage.

****

11. Initiate Draindown Of Affected S/G(s):

    - Establish normal S/G blowdown

      OR

    - Use the S/G blowdown/wet layup system

      OR

    - Use other appropriate drain paths

12. Reset SPDS AND Return To Procedure And Step In Effect

- END -

Progress Energy

NON POM

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

FRP-H.3-BD

FRP-H.3 BASIS DOCUMENT

REVISION 10

DISCUSSION (From the WOG FR-H.3 Basis Document)

1. INTRODUCTION

Guideline FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, provides guidance to address a not satisfied condition (i.e., YELLOW priority) for secondary system inventory that results from a high level condition in any SG. Since the Optimal Recovery Guidelines (ORGs) address the restoration and maintenance of SG narrow range levels following a reactor trip condition, guideline FR-H.3 is considered a YELLOW priority.

Guideline FR-H.3 has been developed and structured to maintain secondary heat sink control, and to provide the utility with an appropriate criterion at which to address the steam generator overfill concern and its implications. The actual evaluation of steam generator overfill status and all of the actions necessary to restore an overfilled SG to service are not addressed in guideline FR-H.3. Steam generator overfill and its potential consequences are the subject of Nuclear Regulatory Commission (NRC) Generic Letter 81-28, dated July 31, 1981 (Reference 1). This letter requested utilities to evaluate credible plant-specific scenarios and include appropriate information in plant-specific training programs stressing the possible consequences of steam generator overfill.

The objective of guideline FR-H.3 is to prevent the levels in the SGs from increasing above the narrow range span so that each SG remains effective for secondary heat removal. In addition, this guideline utilizes the margin between the SG high-high level signal and the top of the narrow range span to prevent an overfill condition. Effectiveness of all steam generators is necessary to permit the operator to optimally control secondary heat sink in responding to a plant emergency.

There are two entries into guideline FR-H.3. One is from a YELLOW priority on the Heat Sink Critical Safety Function Status Tree, F-0.3, based on operator judgement. The other entry is from FR-H.2, RESPONSE TO STEAM GENERATOR OVERPRESSURE, Step 3, if the affected SG narrow range level is high.

Guideline FR-H.3 may be exited at several locations. When all actions of this guideline have been completed, or after it has been determined that SG narrow range levels are below the upper tap and decreasing, the operator is instructed to return to the guideline and step that was in effect when FR-H.3 was entered. In addition, if the affected SG(s) has abnormal radiation indications, the operator is directed to E-3, STEAM GENERATOR TUBE RUPTURE, unless an E-3 or ECA-3 series guideline is in effect, in which case the operator returns to that guideline.

2. DESCRIPTION

SG high level can constitute an initiating event that results in a reactor trip or can occur in combination with other plant conditions following a reactor trip.

Following a reactor trip, either the reactor trip (P-4) signal in combination with a low Tavg signal or an SI signal will isolate main FW flow to the SGs. Depending on the plant condition, the auxiliary feedwater (AFW) system is manually or automatically actuated to establish AFW flow to the steam generators consistent with core decay heat removal requirements.

Following actuation of FW isolation, the operator verifies in E-0, REACTOR TRIP OR SAFETY INJECTION, that appropriate FW control and isolation valves are closed. This ensures that the SGs will not overfill due to excessive main FW addition. The operator then controls AFW flow to restore and maintain the required SG narrow range level. Through verification of FW isolation and control of AFW flow, guideline FR-H.3 is compatible with the Optimal Recovery Guidelines in the restoration and maintenance of SG narrow range level.

A steam generator high level condition may also occur due to a steam generator tube rupture. The recovery technique utilized in guideline FR-H.3 determines if a SGTR exists and then transfers the operator to the E-3 or ECA-3 series guideline as appropriate.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-H.3 is to respond to a SG high level condition and to address the SG overfill concern.

The following subsections provide a summary of the major categories of operator actions and the key utility decision points for guideline FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-H.3 is given in the form of major action categories. These are described below in more detail.

MAJOR ACTION CATEGORIES IN FR-H.3

- o Isolate Affected SG
- o Check Affected SG Radiation
- o Establish Blowdown from the ffected SG

o Isolate Affected SG

After identifying the affected SG, the operator should verify that the main FW pumps are stopped and main FW to the affected SG is isolated. This ensures that no main FW addition will cause the SG level to increase further.

The operator isolates AFW flow to the affected SG to minimize the level increase in the affected SG. With main FW and AFW isolated, no other normal source of water addition is available. The operator should then check the affected SG level to determine if level is still in the narrow range. If level is still in the narrow range, the operator should control AFW flow to decrease the affected SG level. If the operator is successful in reducing level by throttling AFW flow, the operator is transferred to the guideline and step in effect.

If the affected SG level either fails to decrease after AFW isolation or increases above the narrow range, the operator should proceed to isolate the affected SG steam release paths. If level increases above the narrow range, the operator cannot monitor level and cannot ensure that the SG does not overfill. If level fails to decrease, this may be an indication of either isolation valve leakage or steam generator tube rupture.

o Check Affected SG Radiation

Having isolated the affected SG, the operator should evaluate if the affected SG is ruptured by checking its radiation level. If the affected SG is ruptured, the operator is transferred to an E-3 or ECA-3 series guideline, as appropriate.

o Establish Blowdown from the Affected SG

If the affected SG radiation levels are normal, blowdown is established to reduce the level in the affected SG into the normal operating band.

3.2 Key Utility Decision Points

There is one key utility decision point in this guideline when the utility must determine an appropriate course of action. After it has been determined that SG narrow range level has increased to greater than a value corresponding to SG level at the upper tap, an evaluation should be made for SG overfill considerations. The utility must address the methods to be used for the SG overfill evaluation and the actions necessary to return an overfilled SG to service. Guideline FR-H.3 alerts the operator of the need for an evaluation and steam release restrictions, but does not provide the specifics for the evaluation or subsequent actions to address an overfilled SG. The NRC Generic Letter 81-28 (Reference 1) provides information and requirements for utilities to develop the necessary actions to respond to an overfilled SG.

STEP SPECIFIC DESCRIPTION AND RNP DIFFERENCES

The following pages will provide the RNP step number, the ERG step number, the WOG basis for each step where applicable, the differences between the ERG and RNP step, and the Category of deviation (SSD).

RNP STEP	WOG STEP	BASIS/DIFFERENCES
PEC	PEC	WOG BASIS N/A, there is no WOG basis description for the PEC, other than the general description. RNP DIFFERENCES/REASONS The RNP entry conditions include entry from FRP-H.1. WOG FR-H.4 contains a caution pertaining to steam release from a S/G with high level. This caution has been incorporated in FRP-H.4 as a step in order to eliminate action steps in cautions and notes. As part of the step, a transition to FRP-H.3, was added since the caution dealt with high level and FRP-H.3 is the procedure for high level. SSD DETERMINATION This is an SSD per criterion 10 and 11.
C1	C1	WOG BASIS PURPOSE: To alert the operator to the potential of overfilling the steam generator to the point where water may have entered the steam lines BASIS: If the affected SG level has increased above the narrow range, the operator cannot be sure if the SG is filled to the steamline. The objective of the status evaluation is to determine if water is in the steamline. Just decreasing affected SG level into the narrow range does not ensure that water does not remain in the affected SG steamline. An evaluation of the steamline conditions should occur prior to releasing steam from any SG with level above (M.08)% [(M.09)% for adverse containment] to prevent potential damage to piping, valves, or turbines. KNOWLEDGE: Understanding of potential effects of SG overfill, including: • Valve inoperability due to effects of water or two-phase flow • Water hammer effects on main steamlines • Increased dead weight placed on the main steamline and its supports RNP DIFFERENCES/REASONS The RNP caution has been reworded and part of the caution moved to step 2 in order to eliminate action steps within the cautions and notes. SSD DETERMINATION This is an SSD per criterion 11.
N1	N1	WOG BASIS PURPOSE: To define the terminology used in the guideline BASIS: The definition of the word "affected SG" reduces descriptive requirements throughout the remainder of the guideline. RNP DIFFERENCES/REASONS There are essentially no differences.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

SSD DETERMINATION

There are essentially no differences.

1 1 WOG BASIS

PURPOSE: To identify the specific SG that is affected

BASIS:

If the operator confirms that any steam generator narrow range level is above the high-high level feedwater isolation setpoint, he has identified the affected SG and continues in guideline FR-H.3. If all steam generator levels are less than this value, there is no affected SG and the operator is transferred to the guideline in effect. The steam generator high-high level feedwater isolation setpoint is selected for entry into guideline FR-H.3 since steam generator level should always be controlled below this value.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

2 C1 WOG BASIS

See above.

RNP DIFFERENCES/REASONS

The RNP step has included an action step to check S/G level for an overfilled condition. This was removed from the ERG caution. The RNP procedure places the caution or note in an action step to prevent actions within cautions and noted as required by the writer's guide.

SSD DETERMINATION

This is an SSD per criterion 11.

3 2 WOG BASIS

PURPOSE: To verify the automatic FW protective actions of the SG high-high level signal

BASIS:

The operator should immediately verify FW pump trip and FW isolation. If the FW pumps have not tripped, the operator should manually trip them to remove the high pressure source of water to the affected SG. The main feedwater control, bypass and isolation valves should be verified as closed or should be manually closed. This step addresses the possibility that a valve may have failed to automatically close for the automatic feedwater isolation that accompanied the SG high-high level signal.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

4 3

WOG BASIS

PURPOSE: To isolate AFW flow as a potential source of water overfilling the affected SG(s)

BASIS:

AFW flow isolation to the affected SG(s) allows the operator to minimize further level increases. AFW flow can still be maintained to the unaffected SG(s) to control plant conditions.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

5 4

WOG BASIS

PURPOSE: To evaluate the effects of main FW and AFW isolation actions

BASIS:

The operator should continue to monitor affected steam generator narrow range level to determine if level is decreasing. If level is less than the value corresponding to SG level at the upper tap and decreasing, operator actions have been successful. The operator then controls AFW flow to maintain narrow range level in the normal operating band and transfers back to the guideline in effect to continue plant recovery. If level is still above the upper tap value or level is not decreasing, the operator is directed to Steps 5 through 8 where the affected SG(s) is isolated and evaluated for a possible tube failure.

RNP DIFFERENCES/REASONS

There are essentially no differences.

SSD DETERMINATION

This is not an SSD.

RNP STEP	WOG STEP	BASIS/DIFFERENCES
6	5, C6, 6, & 7	WOG BASIS

ERG Step 5 (step 6.a)

PURPOSE: To prevent steam release through the SG PORVs when the affected SG level(s) is above the upper tap

BASIS:

The setpoint should be greater than no-load pressure in order to minimize the steam releases from the affected SG and less than the minimum safety valve setpoint to prevent lifting of the code safety valves. The 25 psi margin is a typical value to allow for opening of the PORV prior to lifting of the safety valve.

KNOWLEDGE:

N/A

ERG C6 (step 6.c)

PURPOSE: To warn the operator that the steamline to the turbine-driven AFW pump must not be isolated if it is the only source of feed flow to the steam generators

BASIS:

If the turbine-driven AFW pump is the only operable source of feed flow to the steam generators (i.e.,no other MD AFW pumps or other operable pumps are capable of providing feed flow to the SGs), then isolation of its steam supply line may degrade system conditions and result in a transition to FR-H.1. Therefore, this isolation must not be performed.

ERG Step 6 (step 6.d & 6.e)

PURPOSE: To isolate the affected SG in order to minimize the potential for radioactive steam releases from the affected SG and to minimize the potential release of two-phase flow in the steamlines which could result in damage to piping, valves and turbines.

BASIS:

The operator arrives at this step after main FW and AFW isolation is completed because level is either above the upper tap or is not decreasing. This indicates a possible SG overfill condition and/or tube rupture. Therefore, the SG steam paths are isolated to prevent potential damage to piping, valves and turbines from two-phase flow and to minimize radioactive steam releases if a SGTR exists.

ERG Step 7 (step 6.a)

PURPOSE: To isolate the affected SG in order to minimize the potential for radioactive steam releases from the affected SG and to minimize the potential release of two-phase flow in the steamlines which could result in damage to piping, valves and turbines.

BASIS:

The operator arrives at this step after main FW and AFW isolation is completed because level is either above the upper tap or is not decreasing. This indicates a possible SG overfill condition and/or tube rupture. Therefore, the SG steam paths are isolated to prevent potential damage to piping, valves and turbines from two-phase flow and to minimize radioactive steam releases if a SGTR exists.

RNP DIFFERENCES/REASONS

The RNP step has combined the ERG steps since they all involve S/G isolation. The RNP step includes the caution from ERG C6 to prevent actions within cautions and notes as required by the writer's guide. The RNP step has also included a step for the steam line drains as these are also common valves closed for S/G isolation. The intent of the ERG has not been changed.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

<u>SSD DETERMINATION</u>

This is an SSD per criterion 10 and 11.

7 N/A <u>WOG BASIS</u>

N/A, this step is not in the WOG ERG.

<u>RNP DIFFERENCES/REASONS</u>

This step was added to support the response to Generic Letter 81-28. Analysis of the steam lines may require structural bracing to prevent failure. This will be evaluated by staff personnel.

<u>SSD DETERMINATION</u>

This is an SSD per criterion 10.

N8 8 <u>WOG BASIS</u>

See below.

<u>RNP DIFFERENCES/REASONS</u>

This note has been added as an additional resource for monitoring S/G radiation as noted in the ERG step 8 for plant specific means. The Phase "A" Override Key Switches may be used for drawing S/G samples for the presence of activity.

<u>SSD DETERMINATION</u>

This is an SSD per criterion 4.

8-10 8 <u>WOG BASIS</u>

<u>PURPOSE</u>: To check the affected SG for a SG tube rupture

<u>BASIS:</u>

Steam generator level increasing or above the narrow range with main FW and AFW flow isolated is one symptom of a steam generator tube rupture. In this step another symptom of a steam generator tube rupture is checked, i.e., abnormal radiation level in the SG. If radiation levels are normal, the operator proceeds in guideline FR-H-3. If radiation levels are abnormal, the affected steam generator is determined to be ruptured and a transition to the appropriate Optimal Recovery Guideline (ORG) is made. If the guideline in effect is an E-3 or ECA-3 series guideline, the operator is addressing steam generator tube rupture recovery in the optimal manner and should return to that guideline. If the guideline in effect is not an E-3 or ECA-3 series guideline, the operator should go to guideline E-3, STEAM GENERATOR TUBE RUPTURE, to address a steam generator tube rupture.

<u>KNOWLEDGE</u>:

- How to obtain secondary radiation level readings including signals that may need to be reset
- "Normal" means the value of a process parameter experienced during routine plant operations.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

RNP DIFFERENCES/REASONS

Plant specific means have been included as directed by the ERG. Radiation Monitor R-15 has not been included in the secondary monitors listed since it is expected at this time for the Turbine to be tripped. In addition, the MSIVs and Bypasses for the affected S/G have been closed by previous steps, thus removing a pathway to monitor R-15.

The ERG RNO step has been placed as a separate step in order to provide a listing of all the E-3 & ECA-3 series procedures. The RNP procedure will return the operator to the ECA-3 series steps directly if these procedure were in effect at the time of entry to this procedure.

SSD DETERMINATION

This is an SSD per criterion 4.

C11 N/A WOG BASIS

N/A, this step is not in the WOG ERG.

RNP DIFFERENCES/REASONS

S/Gs have been previously isolated to prevent damage due to the overfill condition. A step to direct evaluation of the overfilled condition prior to commencing steam release is in effect. This caution serves as a reminder that the evaluation must be completed prior to blowdown.

SSD DETERMINATION

This is an SSD per criterion 10.

11 9 WOG BASIS

PURPOSE: To reduce affected SG level into the normal operating range through blowdown

BASIS:

Level must be decreased into the normal operating range prior to restoring the SG to service. SG blowdown can be used to decrease the affected SG(s) level since Step 8 verified a normal radiation level is present.

RNP DIFFERENCES/REASONS

Plant specific means are listed as directed by the ERG.

12 10 SSD DETERMINATION

PURPOSE: To direct the operator to the proper guideline following successful completion of the steps in this guideline

BASIS:

The operator has done everything possible to mitigate the SG high level condition. Therefore, the operator should continue plant recovery operations by returning to the guideline and step that was in effect at the time FR-H.3 was entered.

WOG BASIS

There are essentially no differences.

RNP DIFFERENCES/REASONS

This is not an SSD.

90. 064 G2.4.46 SRO 001

Given the following plant conditions.

- The plant was at 100% power and experienced reactor trip coincident with a loss of the SUT.
- "A" and "B" EDG are supplying power to 480V buses E-1 and E-2.
- APP-010-E5, EDG RM A COOL FAN HI TEMP/OVLD, alarm is received.
- Readings from a calibrated temperature instrument indicate that EDG A room is at 128°F.

Which ONE (1) of the following completes the statements below?

"A" EDG __(1)__ required to be declared inoperable. The actions required due to the alarm and temperature reading is to __(2)__.

A.✓ (1) is NOT

(2) notify the System Engineer to investigate the cause and initiate corrective action IAW APP-010-E5

B. (1) is NOT

(2) shutdown IAW OP-604, Diesel Generators "A" and "B"

C. (1) is

(2) notify the System Engineer to investigate the cause and initiate corrective action IAW APP-010-E5

D. (1) is

(2) shutdown IAW OP-604, Diesel Generators "A" and "B"

The correct answer is A.

A. Correct - Per APP-010-E5, the EDG A must not be declared inoperable unless the room exceeds 130°F. With the high temperature the system engineer must be notified and the cause of the alarm must be investigated and corrective actions initiated.

B. Incorrect. The first part of the distractor is correct. If the EDG was running for testing and the high temperature / alarm was due to a fan breaker tripping then the EDG would be shutdown IAW OP-604. Fan operation is not discussed in the stem of the question. This makes this distractor plausible.

C. Incorrect. The EDG room temperature must exceed 130°F prior to requiring that the EDG be declared OOS. The second part of the distractor is correct.

D. Incorrect - The EDG room temperature must exceed 130°F prior to requiring that the EDG be declared OOS. If the EDG was running for testing and the high temperature / alarm was due to a fan breaker tripping then the EDG would be shutdown IAW OP-604. Fan operation is not discussed in the stem of the question. This makes this distractor plausible.

Question 90
Tier/Group 2/1
K/A Importance Rating - RO 4.2 SRO 4.2

Emergency Diesel Generator: Ability to verify that the alarms are consistent with the plant conditions.

Reference(s) - Sim/Plant design, APP-010, OST-401-2, OP-604
Proposed References to be provided to applicants during examination - None
Learning Objective - EDG06
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments - K/A match because candidate know the high temperature limit for EDG operability.

SRO: Candidate must assess plant conditions and then select a procedure to mitigate the conditions given. In this situation the SRO must know the content of APP-010-E5.

91. 011 G2.2.38 SRO 001

Given the following plant conditions:

- Plant is in Mode 3.
- PZR level transmitter LT-459 fails LOW.
- No operator actions are taken.
- At 1700 PZR level is 85.1% and rising at a rate of 0.5%/min.
- PZR pressure is being maintained constant.

Which ONE(1) of the following completes the statements below?

At the current rate of level rise __(1)__ is the EARLIEST time at which the ITS 3.4.9, Pressurizer, will NOT be met.

The bases for the limit in this Mode is to ensure that the __(2)__.

A. (1) 1712

(2) RCS does NOT go solid when criticality is achieved and preserves a steam space for pressure control

B.✓ (1) 1714

(2) RCS does NOT go solid when criticality is achieved and preserves a steam space for pressure control

C. (1) 1712

(2) PZR level remains within the calibrated level range and provide protection against water relief through the PZR PORVs

D. (1) 1714

(2) PZR level remains within the calibrated level range and provide protection against water relief through the PZR PORVs

The correct answer is B.

A. Incorrect. The time given is based on the Pressurizer Water Level - High reactor trip setpoint of 91%. The second half of the answer is correct.

B. Correct.

C. Incorrect. The time given is based on the Pressurizer Water Level - High reactor trip setpoint of 91%. The last half of the second distractor is from the basis for the Pressurizer Water Level-High reactor trip with the exception that it is to provide protection against water relief through the pressurizer safety valves. The PZR level indication is calibrated throughout the full range of indication.

D. Incorrect. The first half of the distractor is correct. The last half of the second distractor is from the basis for the Pressurizer Water Level-High reactor trip with the exception that it is to provide protection against water relief through the pressurizer safety valves. The PZR level indication is calibrated throughout the full range of indication.

Question 91
Tier/Group 2/2
K/A Importance Rating - RO 3.6 SRO 4.5

Pressurizer Level Control System (PZR LCS): Knowledge of conditions and limitations in the facility license.

Reference(s) - Sim/Plant design, System Description, ITS 3.4.9, ITS 3.4.9 Bases
Proposed References to be provided to applicants during examination - None
Learning Objective - PZR 010, 012
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.6 / 45.4
Comments - .

SRO: Knowledge of TS bases.

the CV.

The PRT is normally filled to 70% with primary water. A 3 psig nitrogen atmosphere is maintained in the PRT to blanket the water. Primary water may be added to the tank by use of the primary water pumps and valves operated from the RTGB. Water may also be drained from the tank by utilizing either of the RCDT pumps and valves operated from the WDBRS panel.

Steam discharged to the PRT from the PZR PORVs and Safeties is directed to the sparger, a pipe containing spray nozzles, near the bottom of the PRT. This allows the high energy steam to be quenched in the water of the PRT. This will allow limited discharge of steam to the PRT before the pressure in PRT raises sufficiently to rupture the rupture discs.

4.0 INSTRUMENTATION

4.1 PZR Instrumentation

4.1.1 Temperature Instrumentation

The following temperature elements provide indication and alarm on the RTGB:

1. PZR Liquid Space (TE-453)
2. PZR Steam Space (TE-454)
3. PZR Spray Line (TE-451 and 452)
4. PZR Surge Line (TE-450)
5. Discharge of PORV's (TE-463) and each Safety Valve (TE-465, 467, and 469)

4.1.2 Level

Three PZR level transmitters, calibrated at normal operating temperatures, are used to provide signals for reactor protection (High Level Trip).

1. LT-459
2. LT-460
3. LT-461

One PZR level signal, LT-462, is provided for indication when the system is in cold condition and therefore is calibrated at cold conditions.

Channels 459, 460, and 461 are used in protection and are available for control functions by a switch on the RTGB. Normally, Channels 459 and 460 are used for control, and either can be replaced by Channel 461. Channel 459 normally provides signal for charging pump speed control, set point deviation alarm, and will turn on backup heaters (if they are in automatic) on a high level error signal to heat incoming water. Either control channel can provide a letdown isolation signal (shut 460A and B in the CVCS system) and turn off all PZR heaters.

There is an additional PZR level signal LT-607D used for indication on the Dedicated Shutdown panel.

4.1.3 Pressure

There are eight pressure transmitters on the PZR.

1. PT-445

   - provides a signal for operating power operated relief valve PCV-456.
   - provides a high and low pressure alarm.
   - provides pressure indication on the RTGB.

2. PT-444

   - provides pressure indication on the RTGB and at the motor driven auxiliary feedwater pump station.
   - provides signal to a proportional - plus - reset controller (PC-444J) on the RTGB.
   - provides signal to PC-444J for operating power operated relief valve PCV-455C.
   - provides signal to PC-444J for spray valves PCV-455A and B.
   - provides signal to PC-444J for heater control and high controller output alarm.

3. PT-455, 456, and 457

   Are the three PZR pressure protection channels that supply the following to reactor protection and safeguards:

   - High PZR Pressure Trip. (2376 psig)
   - Low PZR Pressure Trip. (1844 psig) – rate sensitive
   - Low Pressure Signal for Safety Injection. (1715 psig)
   - Block permissive/auto unblock of low pressure and steam line DP safeguards and block power operated relief valve (PORV) opening.
   - Input to over temperature Delta-T Reactor Trip and Turbine Runback.

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 11)

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 360°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.

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 10)

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 575.9°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 de-energize and any backup heater in manual would remain energized.

5.1.6 PZR Level Control Setpoints (PZR Figure 9)

1.	Level program as function of Median Select T_{avg} (TM-459)	
	for T_{avg} 547°F	22.2% of level span
	for T_{avg} 575.9°F	53.3% of level span
	(Program is linear from 547°F to 575.9°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

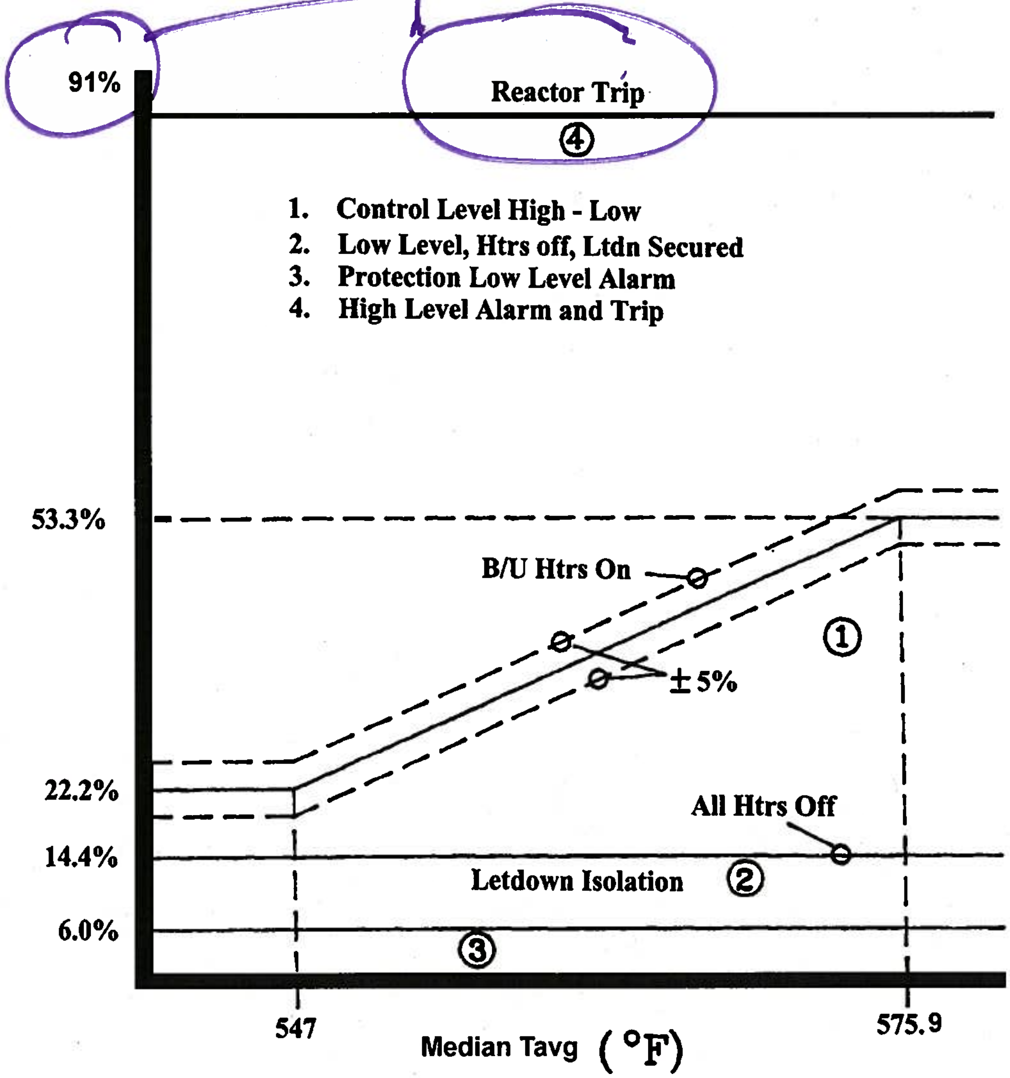
Insurge of RCS Coolant - produced by increase in T_{avg}. An insurge of coolant will reduce volume of the steam bubble causing an increase in the temperature and pressure of the steam. The steam space or bubble becomes superheated and some minor condensation occurs at surface and on walls.

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 LEVEL
LEVEL SETPOINTS
PZR-FIGURE-9

Distractor

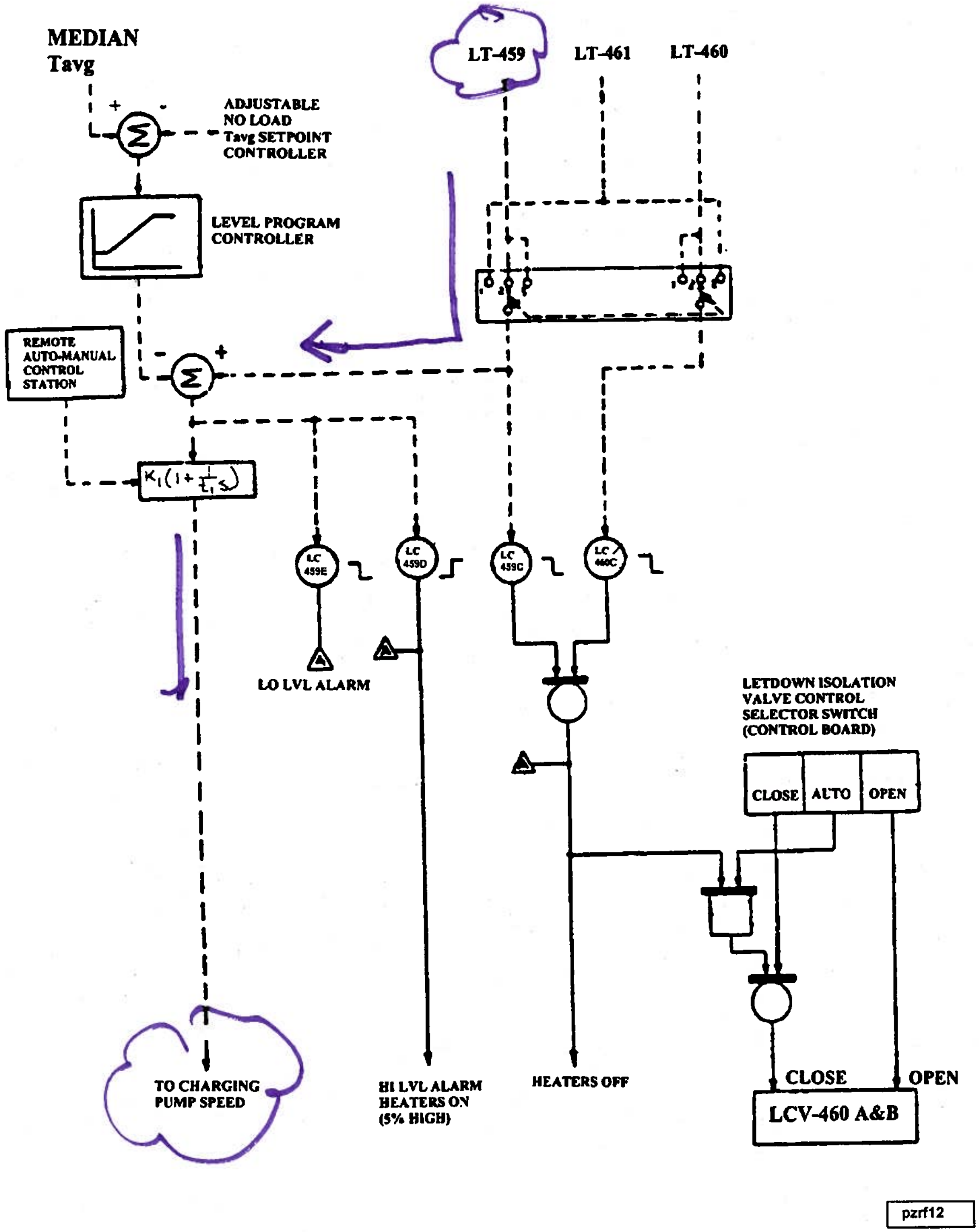


pzrf11

INFORMATION USE ONLY

LEVEL CONTROLLER

PZR-FIGURE-10



INFORMATION USE ONLY

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)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to

(continued)

BASES

BACKGROUND (continued)	a loss of single phase natural circulation and decreased capability to remove core decay heat.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The maximum pressurizer water level limit satisfies Criterion 2 of the NRC Policy Statement. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.
LCO	The LCO requirement for the pressurizer to be OPERABLE with a water level of 63.3% in MODE 1, and ≤ 92% in MODE 2 and MODE 3, ensures that a steam bubble exists. The pressurizer water level of ≤ 63.3% in MODE 1 is the normal programmed level plus 10%, which is consistent with the assumptions used in the accident analyses. The water level of ≤ 92% in MODE 2 and MODE 3 is protected by the pressurizer high level trip setpoint at 91%, and is adequate protection for the pressurizer when load rejection is not a concern. A higher water level is necessary in the pressurizer during cooldown to maintain pressurizer cooldown limits. This level requirement also assures the RCS does not go solid when criticality is achieved. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential

(continued)

92. 045 A2.11 SRO 001

Given the following plant conditions:

- The plant was at 100% RTP.
- Power Distribution Control target values (100%).
  N-41: -2.0% N-43: -2.0%
  N-42: -2.0% N-44: -2.0%

- Target Band +/- 5%
- APL = 102%
- At 1600 the plant has just experienced two spurious OT Delta T turbine runbacks.
- At 1602 the plant stabilized at 88% RTP.

Time	N-41AFD	N-42 AFD	N-43 AFD	N-44 AFD
1603	-10.1	-10.2	-10.1	-10.3
1617	- 8.9	-9.0	-8.9	-9.1
1642	- 7.1	-7.2	-7.1	-7.3
1705	- 6.9	-7.0	-6.9	-7.1
1710	- 6.5	-6.6	-6.5	-6.6

Which ONE (1) of the following completes the statement below?

IAW ITS 3.2.3, Axial Flux Difference, reactor thermal power __(1)__ and the basis for this action is __(2)__.

(REFERENCE PROVIDED)

A. (1) is limited to a maximum of 90% RTP

(2) the xenon axial distribution at this power level is not a significant accident analysis parameter

B. (1) is limited to a maximum of 90% RTP

(2) the radial xenon peaking factors assumed in the accident analysis cannot be exceeded at this power level

C✓ (1) must be reduced to below 50% RTP

(2) the xenon axial distribution at this power level is not a significant accident analysis parameter

D. (1) must be reduced to below 50% RTP

(2) the radial xenon peaking factors assumed in the accident analysis cannot be exceeded at this power level

The correct answer is C.

A. Incorrect. The penalty points have exceeded 1 hour. Per ITS 3.2.3 power must be reduced to below 50%. Power could have stayed between 50% and 90% had less than 1 hour of penalty points been accumulated. The second half of the answer is correct.

B. Incorrect. The penalty points have exceeded 1 hour. Per ITS 3.2.3 power must be reduced to below 50%. Power could have stayed between 50% and 90% had less than 1 hour of penalty points been accumulated. The second half of the distractor is similar to the basis for being outside the acceptable operation limits. Distractor also refers to radial xenon peaking factor vice axial peaking factors.

C. Correct.

D. Incorrect. The first half of the distractor is correct. The second half of the distractor is similar to the basis for being outside the acceptable operation limits. Distractor also refers to radial xenon peaking factor vice axial peaking factors.

Question 92
Tier/Group 2/2
K/A Importance Rating - RO 2.4 SRO 2.9

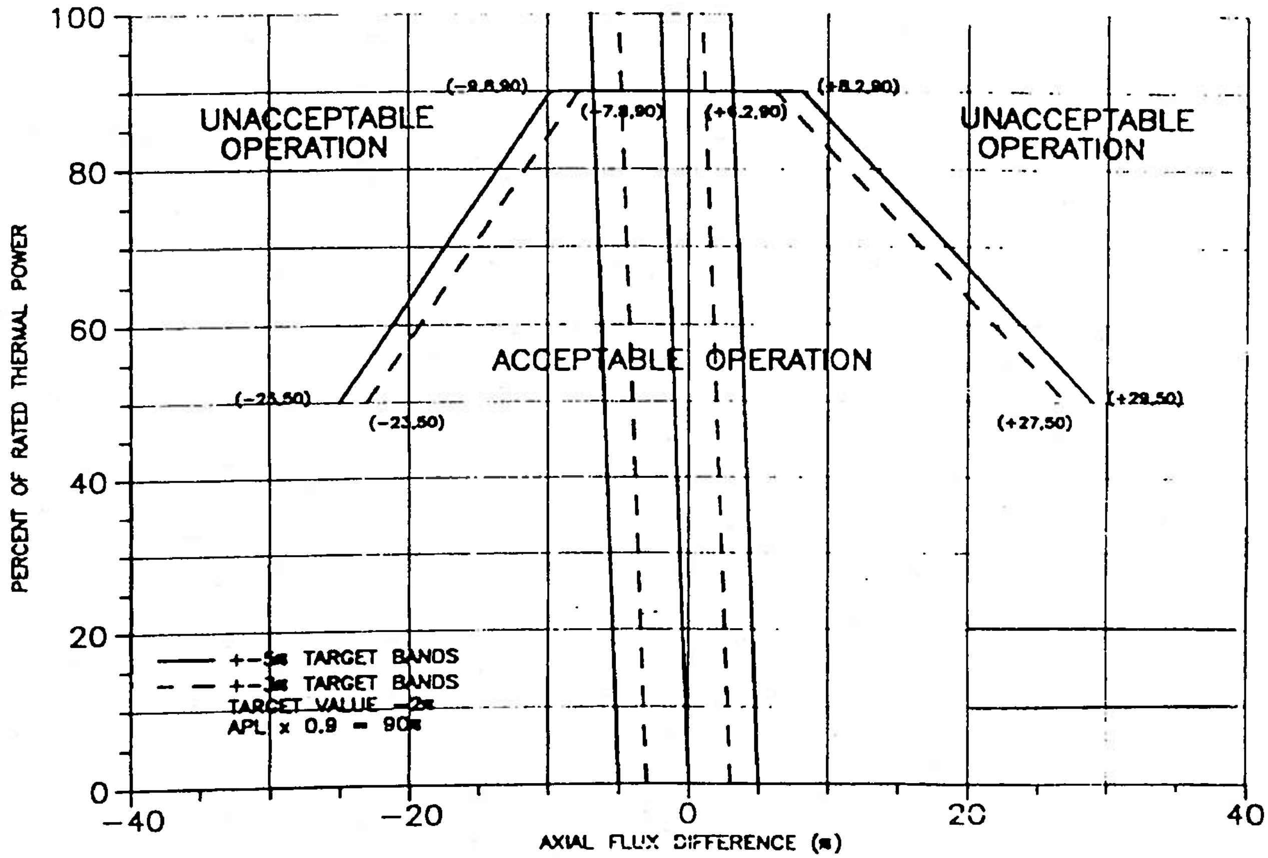
Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Control problems in primary, e.g., axial flux imbalance; need to reduce load on secondary

Reference(s) - Sim/Plant design, ITS 3.2.3, FMP-009, ITS 3.2.3 Bases
Proposed References to be provided to applicants during examination - FMP-009 Target and Operating Band Diagram
Learning Objective - AOP-15-004, FMP-009-007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.5
Comments -

SRO: Application of required actions in accordance with Tech. Specs. and bases of Tech. Specs.

TARGET AND OPERATING BAND EXAMPLES

FIGURE 3



REFERENCE

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The allowable values of the target band are specified in the COLR.

----------------------------NOTE----------------------------
The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
----

b. May deviate outside the target band with THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

----------------------------NOTES----------------------------
1. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

2. The Allowable Power Level (APL) is the limitation placed on THERMAL POWER for the purposes of applying the AFD target flux and operational limit curves. The APL is as follows:

   APL = minimum over Z of $(100\%)(F_Q^{RTP}(Z))(K(Z))/F_Q^V(Z)$
----

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

----------------------------NOTE----------------------------
Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

---------------------------NOTE---------------------------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP or 0.9 APL, whichever is less.	15 minutes

(continued)

Distractor

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.1 and C.2 must be completed whenever Condition C is entered. ---------------------	C.1	Reduce THERMAL POWER to < 50% RTP.	30 minutes
	AND		
THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. *OR* THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.2	Restore cumulative penalty deviation time to less than 1 hour.	Prior to increasing THERMAL POWER to ≥ 50% RTP
D. ---------NOTE--------- Required Action D.1 must be completed whenever Condition D is entered. --------------------- Required Action and associated Completion Time for Condition C not met.	D.1	Reduce THERMAL POWER to < 15% RTP.	9 hours

Progress Energy

R
Reference Use

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 6
PART 5

FMP-009

POWER DISTRIBUTION CONTROL

REVISION 17

TABLE OF CONTENTS

SECTION PAGE

1.0 PURPOSE

1.1 To provide instructions and guidance to ensure that the core axial power distribution is maintained within the limits established in Technical Specifications (ITS) LCO 3.2.3, SR 3.2.3.1 and SR 3.2.3.2

1.2 To provide instructions for identifying, monitoring and controlling divergent axial oscillations.

1.3 To satisfy Technical Specification (ITS) SR 3.2.3.2

2.0 REFERENCES

2.1 Technical Specifications (ITS) 1.1, LCO 3.2.3, SR 3.2.3.1, SR 3.2.3.2, SR 3.2.3.3

2.2 FMP-001, Core Operating Limits Report (COLR)

2.3 XN-76-40(A), Exxon Nuclear Power Distribution Control For Pressurized Water Reactors, September 1976

2.4 XN-NF-77-57 and XN-NF-77-57 Supplement 1 (A), Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II, May 1981

2.5 XN-NF-77-57 Supplement 2(A) and XN-NF-77-57 Supplement 2 Addendum 1 (A), Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II, October 1982

2.6 ANF-88-054(P), PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H.B. Robinson Unit 2, July 1988

2.7 EST-003, Incore/Excore Detector Calibration

2.8 SCM-003, Plant Computer Systems Database Control Procedure

2.9 UFSAR 1.5.3, 3.1.1.2.2, 3.1.2.7, 4.3.1.6, 4.3.2.2, 4.3.2.6, 4.3.3.3, 4.4.3.1, 7.2.1.1.2, 7.2.1.1.7, 7.7.1.5, 7.7.1.6

2.10 EC 47211 ERFIS Data Concentrator replacement.

2.11 EC 47160 NSS and BOP Analysis to Support Appendix K Uprate

3.0 RESPONSIBILITIES

3.1 Reactor Engineering is responsible for performing flux maps every 31 Effective Full Power Days (EFPDs) to determine the Target Axial Flux Difference (AFD) values, the Target Band values and the Allowable Power Level (APL) value in accordance with Technical Specifications (ITS) SR 3.2.3.3 and LCO 3.2.3. Reactor Engineering is also responsible for ensuring that the Control Room Status Board and the ERFIS CAOC software is updated to reflect the correct Target AFD values, Target Band values and APL value. Reactor Engineering is also responsible for ensuring the ERFIS CAOC software is updated to reflect the correct Incore/Excore calibration constants.

3.2 The Control Operator is responsible for maintaining the AFD within the limits specified in Technical Specifications (ITS) LCO 3.2.3 and SR 3.2.3.1 and the COLR. The Control Operator is also responsible for logging the AFD in accordance with Technical Specification (ITS) SR 3.2.3.2 when the AFD Alarms are inoperable.

3.3 The Superintendent - Shift Operations is responsible for reviewing the Manual AFD Monitoring Log.

4.0 PREREQUISITES

N/A

5.0 PRECAUTIONS AND LIMITATIONS

5.1 Any Technical Specification Required Action regarding reactor power limitations, including the setting of trip setpoints, should be based on a Rated Thermal Power (RTP) of 2339 MW_{th}. A trip setpoint based on a 2300 MW_{th} would satisfy Technical Specification requirements; however, if a trip setpoint is based on an RTP of 2300 MW_{th}, then the actual trip would occurr at a lower indicated power when operating based on a 2339 MW_{th} RTP.

6.0 SPECIAL TOOLS AND EQUIPMENT

N/A

7.0 ACCEPTANCE CRITERIA

N/A

8.0 PROCEDURE

8.1 Definitions

8.1.1 Axial Flux Difference (AFD)
The Axial Flux Difference (AFD) is defined as the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector (ITS 1.1). This parameter is synonymous with Delta Flux, Indicated Flux Difference, ΔI, %ΔI, %Δ Flux and Δq. AFD can also be related to core Axial Offset (AO) using the following equation:

AFD = AO * Power Level/Rated Thermal Power.

AFD relates the power in the top of the core to the power in the bottom of the core as seen by the excore NIS Power Range detectors. A separate AFD value is calculated for each NIS Power Range channel. The equations and ERFIS Point IDs used in calculating AFD for each of the four Power Range channels are shown in ATTACHMENT 10.1. It should be noted that the ERFIS AFD is calculated once per minute and is based on 1 minute average values for V(top), V(bottom) and P.

8.1.2 Target Value (TV)
The Target Value, also known as the Target Flux Difference, is the value of AFD determined in conjunction with the measurement of $F_Q^V(Z)$ under equilibrium conditions within 31 EFPD after each refueling and every 31 EFPD thereafter (ITS SR 3.2.3.3). During startup and power ascension following each refueling, the Target Value may be based on design predictions until equilibrium conditions for long term operation are reached. Like AFD, the Target Value is power dependent (examples of the variation of Target Value with power are provided in ATTACHMENT 10.2). A separate Target Value is calculated for each NIS Power Range channel. The Target Value for a Power Range channel is generally the average ERFIS AFD value recorded for that channel during the course of the flux map. The equations and ERFIS Point IDs used in calculating the power dependent Target Value for each of the four Power Range channels are shown in ATTACHMENT 10.1.

8.1.3 Allowable Power Level (APL)

The Allowable Power Level is the limit placed on reactor power due to the $F_Q^V(Z)$ peaking factor. The APL is used in applying the AFD target flux and operational limit curves (ITS LCO 3.2.3). The equation for determining the APL is provided in Technical Specification (ITS) LCO 3.2.3. The effect of APL on the Target Bands and Operating Bands can be seen in ATTACHMENT 10.2

8.1.4 Target Bands (TB)

The Target Bands establish a region of operation around the Target Value in which the AFD may vary without adversely affecting the axial power distribution or the axial xenon distribution. The allowable values of the Target Band are provided in the COLR for each cycle. The allowable Target Band values are currently ±3% and ±5%. If the APL is less than 90% RTP, then the values of the Target Bands are reduced to account for the effects of the reduced APL; however, they are still referred to as the ±3% and ±5% Target Bands in order to maintain a consistent nomenclature under all operating conditions. Since the Target Band moves with the Target Value, separate Upper and Lower Target Bands are calculated for each NIS Power Range channel (examples of the variation of the Target Bands with power and with APL are provided in ATTACHMENT 10.2). The equations and ERFIS Point IDs used in calculating the Target Bands for each of the four Power Range channels are shown in ATTACHMENT 10.1.

8.1.5 Operating Bands (OB)

The Operating Bands establish a region of acceptable operation outside of the Target Bands in which the AFD may vary for a short time period without adversely affecting the axial power distribution or the axial xenon distribution. The Operating Bands are only applicable to operation at power levels between 50% RTP and 90% RTP (or 90% APL if the APL is less than 100% RTP). The allowable values of the Operating Bands are provided in the COLR for each cycle. The Operating Bands use the same nomenclature as the Target Bands (i.e. ±3% and ±5% Operating Bands). The ±3% Operating Bands are used with the ±3% Target Bands and the ±5% Operating Bands are used with the ±5% Target Bands. If the APL is less than 90% RTP, then the values of the Operating Bands are reduced to account for the effects of the reduced APL; however, they are still referred to as the ±3% and ±5% Operating Bands in order to maintain a consistent nomenclature under all operating conditions. If 0.9*APL is less than 90% RTP, then the top of the Operating Bands is limited to 0.9*APL. The Operating Bands, like Target Bands, move with the Target Value (examples of the variation of the Operating Bands with power and APL are provided in ATTACHMENT 10.2). Separate Upper and Lower Operating Bands are calculated for each NIS Power Range channel. The equations and ERFIS Point IDs used in calculating the Operating Bands for each of the four Power Range channels are shown in ATTACHMENT 10.1.

8.1.6 Penalty Points

Penalty Points are used to track the amount of time that operation outside of the Target Bands has occurred. Penalty point accumulation only occurs when two or more operable Power Range channels indicate that AFD is outside of the Target Bands. Penalty points are related to time of operation outside of the Target Bands through the following relations:

≥50% RTP: 1 Penalty Point= 1 minute outside the Target Bands

<50% RTP: 1 Penalty Point= 2 minutes outside the Target Bands

Penalty points are eliminated by operation within the Target Bands at the same rates at which they were accumulated. The ERFIS Point ID used to track penalty point accumulation is shown in ATTACHMENT 10.1.

8.1.7 Target Warning Bands (TWB)

The Target Warning Bands are provided to alert the Operator that the AFD for one or more NIS Power Range channels is approaching the Target Bands (examples of the Target Warning Bands are provided in ATTACHMENT 10.2). Separate upper and lower Target Warning Bands are calculated for each NIS Power Range channel. The equations and ERFIS Point IDs used in calculating the Target Warning Bands for each of the four Power Range channels are shown in ATTACHMENT 10.1.

8.1.8 Operating Warning Band (OWB)

The Operating Warning Bands are provided to alert the Operator that the AFD for one or more NIS Power Range channels is approaching the Operating Bands (examples of the Operating Warning Bands are provided in ATTACHMENT 10.2). Separate upper and lower Operating Warning Bands are calculated for each NIS Power Range channel. The equations and ERFIS Point IDs used in calculating the Operating Warning Bands for each of the four Power Range channels are shown in ATTACHMENT 10.1.

8.2 Power Distribution Control Methodology

8.2.1 The Power Distribution Control (PDC) methodology for controlling the core axial power distribution tries to avoid the "building-in" of adverse axial power distributions during plant operations by maintaining a relatively constant power shape based on the equilibrium conditions encountered throughout a given core cycle. Maintaining a relatively constant axial power shape ensures that the $F_Q^V(Z)$ peaking factor limit will not be exceeded. Since the "building-in" of adverse power shapes is a function of power level, the restrictions on deviation from the Target Value also vary with power level.

8.2.2 The Operator can use the RTGB AFD meters or the ERFIS AFD point IDs to monitor the status of AFD. Spurious alarms and any other applicable comments relating to AFD monitoring should be entered in the Control Operator's narrative log, when necessary.

8.2.3 The AFD is considered to be outside of the Target Bands or Operating Bands when the AFD for two operable Power Range channels exceed their Target Bands or Operating Bands.

8.2.4 At power levels ≥90% RTP (or 0.9*APL, whichever is less), the AFD must be maintained within the Target Bands. If the AFD is outside of the Target Bands then the AFD must be returned to within the Target Bands within 15 minutes or power must be reduced below 90% RTP (or 0.9*APL, whichever is less) within 30 minutes (Technical Specification (ITS) LCO 3.2.3 Conditions A and B). Penalty Points are accumulated while the AFD is outside the Target Bands.

8.2.5 At power levels <90% RTP (or 0.9*APL, whichever is less) but ≥50% RTP, the AFD may deviate from the Target Bands as long as the total number of Penalty Points incurred over the previous 24 hours does not exceed 60 Penalty Points. The AFD must be maintained within the Operating Bands. If the number of accumulated Penalty Points exceeds 60 or if the AFD is outside of the Operating Bands then power must be reduced below 50% RTP within 30 minutes (Technical Specification (ITS) LCO 3.2.3 Condition C). If power is not reduced below 50% RTP within 30 minutes then power must be reduced below 15% RTP within 9 hours (ITS LCO 3.2.3 Condition D)

8.2.6 At power levels <50% RTP, the AFD may deviate from the Target Bands since adverse power shapes at lower power levels are sufficiently accounted for by the F(ΔI) input to the Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) Reactor Protection setpoints. A power increase to ≥50% RTP is permitted only if the accumulation of Penalty Points within the previous 24 hour period does not exceed 60 points.

8.2.7 At power levels <15% RTP, AFD monitoring is not required by the Technical Specifications (ITS LCO 3.2.3). The ERFIS CAOC program does not perform any AFD calculations below 15% power in order to prevent extraneous alarms during periods when the instrumentation noise levels are significant and power operations is minimal.

8.2.8 Certain evolutions such as Incore/Excore calibrations require the AFD to be outside of the Target Bands. Deviation from the Target Bands for a limited amount of time is permitted by the Technical Specifications provided the AFD is maintained within the Operating Bands (Technical Specifications (ITS) LCO 3.2.3).

8.3 ERFIS Monitoring of AFD

8.3.1 Normal Operation
During normal operation above 15% power, the ERFIS CAOC software program calculates the following values for each of the four NIS Power Range channels once per minute:

1. AFD;
2. Target Value for the current power level;
3. Upper and Lower Target Bands for the current power level;
4. Upper and Lower Operating Bands for the current power level;
5. Upper and Lower Target Warning Bands for the current power level; and,
6. Upper and Lower Operating Warning Bands for the current power level.

The CAOC software program compares the calculated AFD for each Power Range channel to that channel's Target Warning Bands, Target Bands, Operating Warning Bands and Operating Bands. If a Power Range channel crosses one or more of the bands, the ERFIS CAOC program informs the Operator of the condition by providing a message on the ERFIS alarm screen and a report on the Control Room printer and if necessary by actuating the appropriate annunciators on the RTGB (APP-005-D6, Δ FLUX WARNING/STATUS and APP-005-E4, Δ FLUX ALARM). The ERFIS CAOC reports that are printed on the Control Room printer are retained as part of the operating logs.

8.3.1 (Continued)

The ERFIS CAOC software program automatically accumulates Penalty Points at the appropriate rate based on power level if two or more operable Power Range channels are outside of their Target Bands. The program also automatically subtracts penalty points at the appropriate rate if the AFD is within the Target Band and the required time since the points were accumulated (24 hours) has elapsed.

8.3.2 ERFIS CAOC Alarms
If the calculated AFD for any channel is outside of any of the bands for that channel then an alarm is generated via annunciators APP-005-D6 and/or APP-005-E4. The specific cause of the alarm (i.e. the identification of the channel and the band that has been exceeded) is displayed on the ERFIS alarm screen and a report is printed on the Control Room printer. Examples of the types of warning/status/alarms messages are shown in ATTACHMENT 10.3.

8.3.3 ERFIS CAOC Shift Summary Report
A Shift Summary Report is printed on the Control Room printer at 0730, 1530, and 2330 hours. The Shift Summary Report is intended to provide information on the change in AFD with time to allow the Operator or Reactor Engineer to easily detect axial oscillations or instrument calibration problems.

The Shift Summary Report lists the AFD for each of the four channels at the time of the report as well as the Minimum AFD and Maximum AFD each of the channels reached during the eight hour period preceding the report. The report also lists the values for the Target Bands, Target Warning Bands, Operating Bands and Operating Warning Bands for each channel at the time of the report. The report has blanks for the Operator to record the AFD indication from the RTGB meters in order to verify that the RTGB indicated AFD is within 2% of the ERFIS calculated AFD. A deviation between the RTGB indicated AFD and the ERFIS calculated AFD of greater that 2% is not expected and would warrant further investigation to determine if an ERFIS or NIS problem exists. The Shift Summary Report is retained as part of the normal operating logs.

8.3.4 Removing a Channel From Service in CAOC

A NIS Power Range channel may be removed from service in the ERFIS CAOC program by removing that channel's instantaneous % power point from scan. This will generate an APP-005-D6, Δ FLUX WARNING, alarm with the following description displayed on the CAOC alarm report:

CHANNEL #__ NOW OUT OF SERVICE.

A Power Range channel may be returned to service in the ERFIS CAOC program by restoring that channel's instantaneous % power point to scan. This will generate an APP-005-D6, FLUX STATUS, alarm with the following description displayed on the CAOC alarm report:

CHANNEL #__ JUST RETURNED TO SERVICE.

The ERFIS point IDs used in removing channels from service and restoring channels to service are as follows:

NIS CHANNEL	CAOC CHANNEL	ERFIS POINT ID
N41	#1	NIN0041A
N42	#2	NIN0042A
N43	#3	NIN0043A
N44	#4	NIN0044A

8.3.5 Demand AFD Report Disabling Alarms and Penalty Logging
During certain activities such as Incore/Excore calibrations or other maintenance or calibration activities involving the Power Range channels, it may be necessary to demand an AFD printout, disable the alarm function and/or Penalty Point logging function of the ERFIS CAOC computer program to provide immediate information or to prevent extraneous alarms and/or inadvertent accumulation of Penalty Points.

To initiate any of these ERFIS functions select TOC AFD4. The current state of these functions is reflected in the "Current State" column in the lower part of the display. Select the button appropriate for the desired function to toggle the desired state in the "New State" column. The "New State" indication will update in the lower part of the screen to reflect the selected condition. Toggle the buttons as appropriate to enable or disable logging and alarming or to demand an AFD printout. Once the desired states have been input, select the "Update". Button to execute the new states. The AFD demand print will only print once and the CAOC program will set the "Current State" to Off automatically. The Alarm and logging states will remain until new states are input and Update Button is selected. Disabling of the Alarm or Penalty Point Logging functions is treated as removing the AFD monitoring program from service and manual monitoring of AFD is required in accordance with Technical Specification (ITS) SR 3.2.3.2. An alternative to disabling the Alarm and Penalty Point Logging functions during Incore/Excore calibrations and other maintenance activities is to allow the Penalty Points to be accumulated and remove any invalid points after completion of the calibration as described in 8.3.7.

8.3.6 ERFIS Out Of Service
If the ERFIS computer, the CAOC software program or the RTGB annunciators which provide AFD monitoring (APP-005-D6 or APP-005-E4) are inoperable then manual monitoring of AFD is required in accordance with Technical Specification (ITS) SR 3.2.3.2. Manual monitoring is described in Section 8.4.

8.3.7 ERFIS Computer Restart

Whenever the ERFIS computer is restarted, the Operator should verify that the Penalty Point "file" is accurate and should manually update the file if necessary. Several options are available to perform this verification/updating.

1. If no Penalty Points had been accumulated prior to ERFIS being out of service and no Penalty Points were accumulated while ERFIS was out of service, then check that ERFIS Point ID NPU0941 reflects 0 Penalty Points or, if desired, perform the following to view the contents of the Penalty Time Buffer:

    a. Type in the turn on code "CAOCEDIT" or select the TOC from the ERFIS NSSS menu.

    b. Enter the desired beginning time to start browsing at the selected time in the Penalty Point Buffer.;

    c. Use the "-Time", "+Time", "-1 hour", "+1 hour" buttons to sequence forward and backward through the Penalty Point Buffer;

    d. Select "Exit" when finished.

2. If invalid Penalty Point data is present in the Penalty Point buffer and no valid Penalty Points have been accumulated in the previous 24 hours then the entire 24 hour Penalty Point buffer may be initialized to "No Penalty" by performing the following:

    a. Type in the turn on code "AFD1";or select the TOC from the ERFIS NSSS menu.

    b. Click on the initialization box and then select the "initialize" button to begin clearing the penalty point buffer.

    c. When the confirmation message is displayed select "yes" to continue or "no". If "yes" is selected, a confirmation message that the penalty point buffer has been cleared will be displayed and the program will automatically exit. If "no" then select "Exit" to close the window.

8.3.7 (Continued)

3. If valid Penalty Points had been accumulated prior to ERFIS being out of service but no Penalty Points were accumulated while ERFIS was out of service and ERFIS was out of service for less than 24 hours then perform the following to retain the earlier valid Penalty Point data and insert "No Penalty" into the buffer for the period of time that ERFIS was out of service:

   a. Type in the turn on code "AFD2";or select the TOC from the ERFIS NSSS menu

   b. Enter the time that ERFIS went out of service as the "Restart Begin Time"

   c. Enter “0” in the “Restart penalty rate” and then select the “Restart” button.

   d. When the confirmation message is displayed select “yes” to continue or “no”. If “yes” is selected, a confirmation message that the Penalty point Buffer” has been restarted will be displayed and the program will automatically exit. If “no” then select “Exit” to close the window.

4. If valid Penalty Points were accumulated while ERFIS was out of service then perform the following to manually update any or all of the 24 hour Penalty Time buffer:

   a. Type in the turn on code "CAOCEDIT";or select the TOC from the ERFIS NSSS menu.

   b. Using data from ATTACHMENT 10.5, enter the time at which the penalty update block is to begin;

   c. Using data from ATTACHMENT 10.5, enter the time the penalty update block is to end;

8.3.7.4 (Continued)

d. Using data from ATTACHMENT 10.5, enter the appropriate penalty rate for the update block from the following choices:

PENALTY VALUE	PENALTY RATE	POWER
0	No Penalty	Any
1	½ Penalty Point per minute	<50%
2	1 Penalty Point per minute	>50%

e. If desired, the penalty buffer can be updated on a minute by minute basis with the "CAOCEDIT" TOC, ELSE go to step f.

1) Type in the turn on code "CAOCEDIT" or select the TOC from the ERFIS NSSS menu.

2) Enter the desired beginning time to start browsing at the selected time in the penalty point buffer;

3) Use the "-Time", "+Time", "- hour" "+ hour" buttons to sequence forward and backward through the penalty point buffer.

4) Enter the desired penalty rate for the desired hour/minute as needed using data from Attachment 10.5 and the above penalty point value table used in step d.

f. When finished with a block or individual hour/minute update, validate the updated penalty time entries by browsing through the buffer as describded in step g above. If the penalty point data is correct, then select the "Update Buffer" button to incorporate the update.

g. Select the "Exit" button to close the window

8.3.8 ERFIS CAOC Program Constants
Except for the ERFIS Point IDs described above which are provided for the Operator's use, all changes to CAOC program constants should be performed in accordance with SCM-003.

8.4 Manual Monitoring of AFD

8.4.1 If the ERFIS computer, the CAOC software program or the RTGB annunciators which provide AFD monitoring (APP-005-D6 or APP-005-E4) are inoperable then manual monitoring of AFD is required in accordance with Technical Specification (ITS) SR 3.2.3.2.

8.4.2 If the Thermal Power is $\geq$90% RTP or 0.9*APL, whichever is less, then the AFD indicated on the RTGB meters must be verified to be within limits and must be logged on ATTACHMENT 10.5 within 15 minutes of the alarms becoming inoperable and every 15 minutes thereafter.

8.4.3 If the Thermal Power is <90% RTP or 0.9*APL, whichever is less, then the AFD indicated on the RTGB meters must be verified to be within limits and must be logged on ATTACHMENT 10.5 within 1 hour of the alarms becoming inoperable and every 1 hour thereafter.

8.4.4 If two or more operable excore channels indicate that AFD is outside of the Target Band then the applicable Actions of Technical Specifications (ITS) LCO 3.2.3 must be completed and the resultant Penalty Points must be logged on ATTACHMENT 10.5 in accordance with the following rates:

POWER LEVEL	PENALTY POINT RATE
≥ 50% RTP	1 Point/minute
<50% RTP	1/2 Point/minute

8.4.5 If the total number of Penalty Points exceeds 60 in a 24 hour period, then the applicable Actions of Technical Specifications (ITS) LCO 3.2.3 must be completed.

8.4.6 Penalty point are eliminated 24 hours after accumulation by operation within the Target Bands at the same rate at which they were accumulated. Elimination of penalty points is denoted by a negative (-) value on ATTACHMENT 10.5.

8.5 Divergent Axial Oscillation Control

NOTE: UFSAR Sections 1.5.3, 3.1.2.7 and 7.7.1.6 state that procedures are available for the Operator to control axial oscillations. This section provides those procedures.

8.5.1 Axial oscillations can be induced by changes in power level, control rod movement, or a combination of both. A sudden change in either will cause some change in flux shape with a resulting redistribution of xenon concentrations in the core. This will invariably start a xenon oscillation in the axial direction.

8.5.2 Axial oscillations are easily started by control rod motion. The flux shape is very dependent on control rod position due to the high neutron absorption of the control rods. Improper use of the control rods at any time can create very serious oscillations. On the other hand, control rods are the best method for manually controlling a xenon oscillation; therefore, use extreme care when making extensive control rod movement. The control bank should be moved slowly and action taken prior to reaching any limit. Attempt to maintain the control rods retracted as far as is practical during steady-state and power ramp conditions to minimize the "pinching" effect of the control bank.

8.5.3 Depending on the core conditions and Operator actions, an axial oscillation can be dampened out or amplified until it becomes divergent. The most serious axial oscillation is the divergent oscillation. Its amplitude increases with each cycle if left to oscillate naturally and can be increased in size by control rod motion if not carefully applied.

8.5.4 The most important factor in control of a divergent oscillation is the timing of the oscillation. The core oscillates axially with a period of 26 hours; that is, it takes 26 hours for the same point on the sinusoidal wave to reoccur (see ATTACHMENT 10.4). The wave may be halved and quartered down to 6.5 hour intervals where the oscillation goes from maximum AFD to Target Value, Target Value to minimum AFD, etc. Once the timing of the oscillation has been determined, the following control procedure may be used to stop or reduce the oscillation to a minimum:

NOTE: ATTACHMENT 10.4 provides an illustration of the control procedure.

If the oscillation has just begun, an accurate timing determination may be difficult. If the Target Value is known accurately, the control rods should be inserted 5 hours after crossing the Target Value in the positive direction since the equilibrium AFD to peak AFD time is 6.5 hours.

If sufficient rod worth is present, the procedure may be performed in reverse by withdrawing control rods 1.5 hours before the minimum AFD.

The control rod insertion is best done slightly early rather than late since if the insertion is done late an amplification might result which is undesirable.

If after returning the control rods to the initial position the oscillation continues upward, then the control rods were returned to normal too soon. If a downward swing starts, then the control rods were left in too long.

A way of approximating the proper insertion time is to consider it 11.5 hours after the last minimum AFD.

Do not allow the power level to change during the control rod maneuvers since this may induce additional unwanted oscillations which will eventually have to be dampened.

If the core is divergent, the least disturbance may become significant after 1 or 2 periods of oscillation. Therefore, if the control procedure does not work effectively, additional control measures should be taken as soon as practical.

The procedure for control of divergent oscillations can also be effectively applied to very large convergent oscillations where there is a possibility of exceeding control rod insertion limits if normal methods were used.

8.5.4 (Continued)

1. At 1.5 hours before the most positive peak, determine the AFD from ERFIS or the AFD meters and insert the control rods using dilution until the AFD reaches the Target Value;
2. Determine the change in AFD caused by the control rod insertion (AFD before insertion - Target Value);
3. Maintain a constant control rod position and allow AFD to decrease;
4. When AFD is lower than the Target Value by the amount determined in Step 8.5.4.2, withdraw control rods to their original position using boration.

9.0 RECORDS

ATTACHMENT 10.5 must be sent to the vault as a QA record

10.0 ATTACHMENTS

10.1 ERFIS CAOC Software And AFD Related Parameters

10.2 Target and Operating Band Examples

10.3 ERFIS CAOC Warning/Status/Alarm Messages

10.4 Divergent Axial Oscillation

10.5 Manual AFD Monitoring Log

ERFIS CAOC SOFTWARE AND AFD RELATED PARAMETERS

EQUATIONS AND ERFIS POINTS USED TO CALCULATE AXIAL FLUX DIFFERENCE (AFD)

$$AFD = \frac{V(top) - V(bottom)}{V(top) + V(bottom)} * K * P$$

where:

- V(top) = Voltage representing the core power as seen by a NIS Power Range channel top detector;
- V(bottom)= Voltage representing the core power as seen by a NIS Power Range channel bottom detector;
- K = Incore/Excore calibration constant determined in accordance with EST-003.
- P = Power Level (%) at the time of the calculation.

Parameter	N41	N42	N43	N44
AFD	NPU0900	NPU0901	NPU0902	NPU0903
V(top)	NIN0051M	NIN0053M	NIN0055M	NIN0057M
V(bottom)	NIN0052M	NIN0054M	NIN0056M	NIN0058M
K	NPK1612	NPK1613	NPK1614	NPK1615
P	NIN0041M	NIN0042M	NIN0043M	NIN0044M

EQUATIONS AND ERFIS POINTS USED TO CALCULATE TARGET VALUES

$$\text{Target Value (P)} = \text{Target Value}_{ref} * P/P_{ref}$$

where:

Target Value (P) = Target Value at power P;
Target Value_{ref} = Target Value during flux map;
P = Current power level; and,
P_{ref} = Power level during flux map.

Parameter	N41	N42	N43	N44
Target Value (P)	N/A	N/A	N/A	N/A
Target Value_{ref}	NPK1603	NPK1604	NPK1605	NPK1606
P	NIN0041M	NINOO42M	NIN0043M	NIN0044M
P_{ref}	NPK1608	NPK1609	NPK1610	NPK1611

ERFIS CAOC SOFTWARE AND AFD RELATED PARAMETERS

EQUATIONS AND ERFIS POINTS USED TO CALCULATE TARGET BAND VALUES

APL $\geq$ 90% RTP — TB Low Limit = TV - TB Value
TB High Limit = TV + TB Value

APL < 90% RTP — TB Low Limit = TV - [TB Value * (APL/100)]
TB High Limit = TV + [TB Value * (APL/100)]

where: TV(P) =Target Value
TB Value = Selected Target Band value ($\pm$3% or$\pm$5%)
APL = Allowable Power Level

Parameter	N41	N42	N43	N44
Target Band Low Limit	NPU0905	NPU0906	NPU0907	NPU0908
Target Band High Limit	NPU0909	NPU0910	NPU0911	NPU0912
Target Band Value	NPK1620	NPK1620	NPK1620	NPK1620
APL	NPK0320	NPK0320	NPK0320	NPK0320

EQUATIONS AND ERFIS POINTS USED TO CALCULATE TARGET WARNING BAND VALUES

TWB Low Limit = TB Low Limit + TWB Value
TWB High Limit = TB High Limit - TWB Value

where: TWB = Target Warning Band; and,
TB = Target Band.

Parameter	N41	N42	N43	N44
TWB Low Limit	NPU0913	NPU0914	NPU0915	NPU0916
TWB High Limit	NPU0917	NPU0918	NPU0919	NPU0920
TB Low Limit	NPU0905	NPU0906	NPU0907	NPU0908
TB High Limit	NPU0909	NPU0910	NPU0911	NPU0912
TWB Value	NPK1622	NPK1622	NPK1622	NPK1622

ERFIS CAOC SOFTWARE AND AFD RELATED PARAMETERS

EQUATIONS AND ERFIS POINTS USED TO CALCULATE OPERATING BAND VALUES

5% OB, APL≥90% RTP: Upper OB = TV + 0.5 * [110 - (P/PL * 100)]
Lower OB = TV - 0.4 * [110 - (P/PL *100)]

5% OB, APL<90% RTP: Upper OB=TV+ {0.5 * [110 -(P/PL*100)]*(APL/100)}
Lower OB= TV - {0.4 * [110 - (P/PL*100)]*(APL/100)}

3% OB, APL≥90% RTP: Upper OB = TV + 0.5 *[106 - (P/PL * 100)]
Lower OB = TV - 0.4 * [105 - (P/PL *100)]

3% OB, APL<90% RTP: Upper OB=TV + {0.5*[106-(P/PL*100)]*(APL/100)}
Lower OB= TV - {0.4 * [105 -(P/PL*100)]*(APL/100)}

where: OB = Operating Band;
TV =Target Value;
P = Average NIS Power Range power expressed as % RTP;
PL = The lesser of 100% RTP or APL; and,
APL = Allowable Power Level

Parameter	N41	N42	N43	N44
Operating Band Low Limit	NPU0921	NPU0922	NPU0923	NPU0924
Operating Band High Limit	NPU0925	NPU0926	NPU0927	NPU0928
P	NIN0041M	NIN0042M	NIN0043M	NIN0044M
APL	NPK0320	NPK0320	NPK0320	NPK0320

EQUATIONS AND ERFIS POINTS USED TO CALCULATE OPERATING WARNING BAND VALUES

OWB Low Limit = OB Low Limit + OW Envelope Value
OWB High Limit = OB High Limit - OW Envelope Value

where: OWB = Operating Warning Band; and,
OB = Operating Band.

Parameter	N41	N42	N43	N44
OWB Low Limit	NPU0929	NPU0930	NPU0931	NPU0932
OWB High Limit	NPU0933	NPU0934	NPU0935	NPU0936
OB Low Limit	NPU0921	NPU0922	NPU0923	NPU0924
OB High Limit	NPU0925	NPU0926	NPU0927	NPU0928
OW Envelope Value	NPK1616	NPK1616	NPK1616	NPK1616

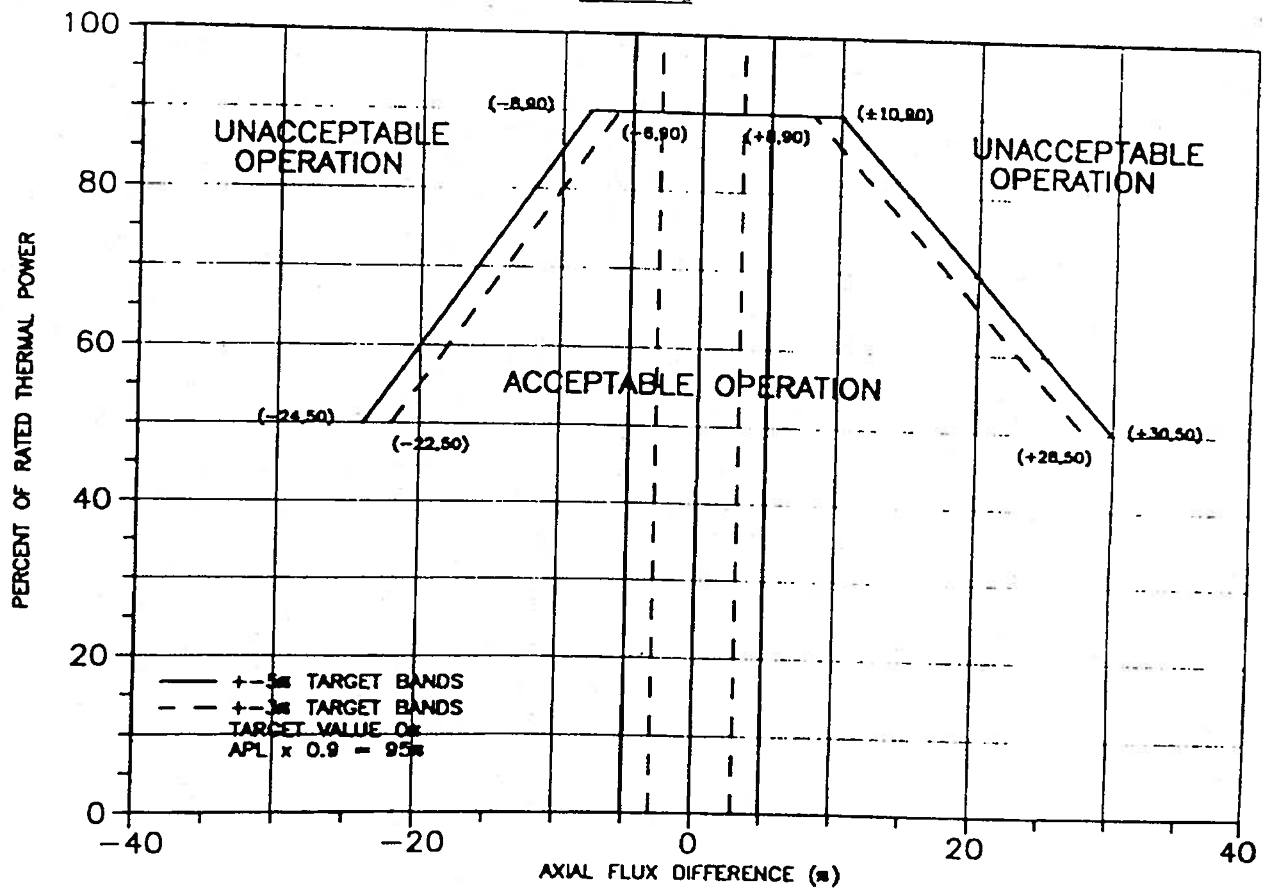
ERFIS CAOC SOFTWARE AND AFD RELATED PARAMETERS

	N41	N42	N43	N44
Minimum AFD over Previous 8 hours	NPU0950	NPU0951	NPU0952	NPU0953
Maximum AFD over Previous 8 hours	NPU0954	NPU0955	NPU0956	NPU0957

Accumlated 24 Hour Penalty Points	NPU0904
Penalty Points at Last Execution of Program	NPU0941
Power Level at Last Execution of Program	NPU0939

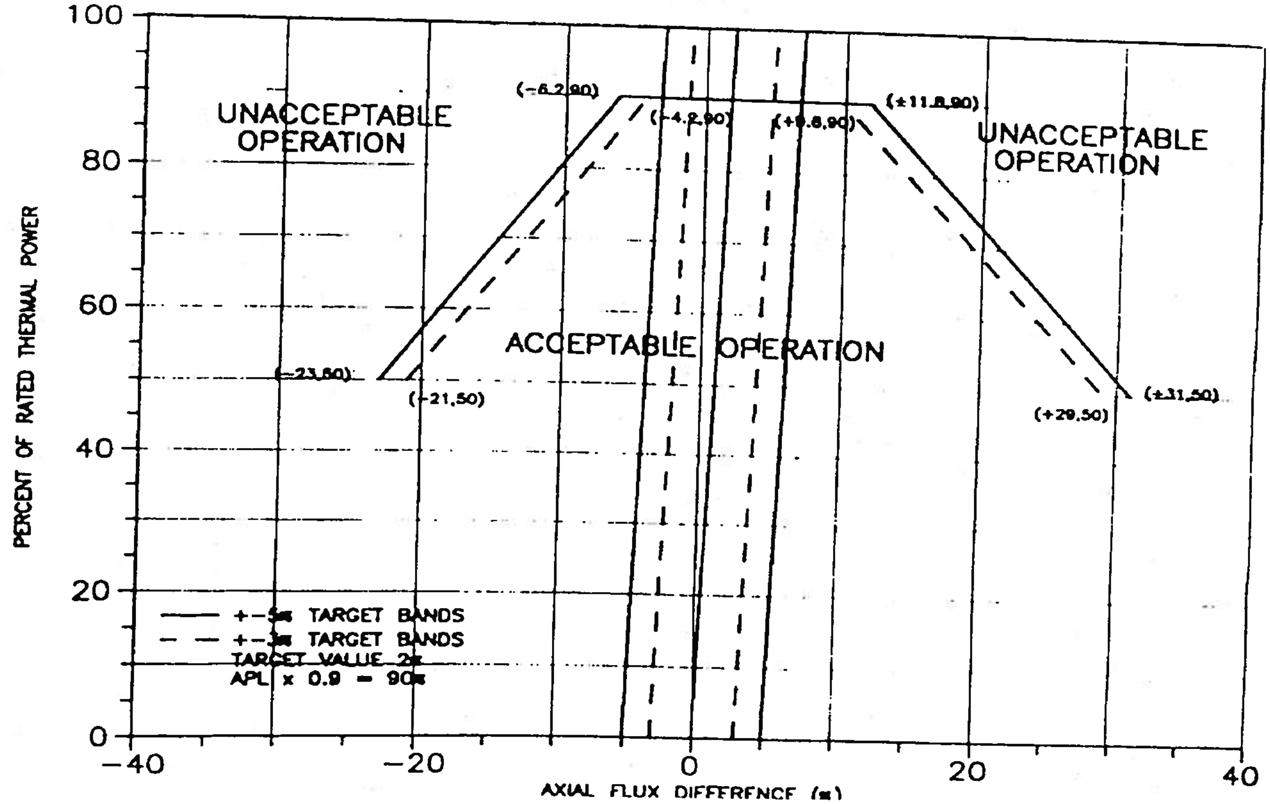
TARGET AND OPERATING BAND EXAMPLES

FIGURE 1



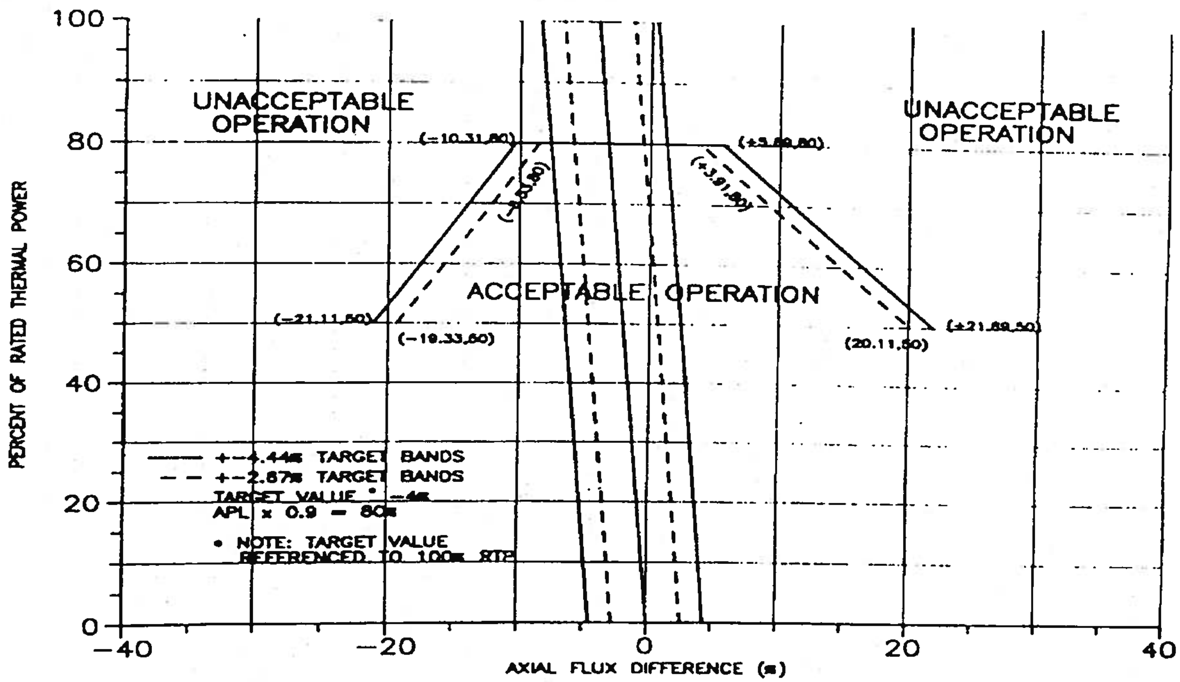
TARGET AND OPERATING BAND EXAMPLES

FIGURE 2



TARGET AND OPERATING BAND EXAMPLES

FIGURE 4



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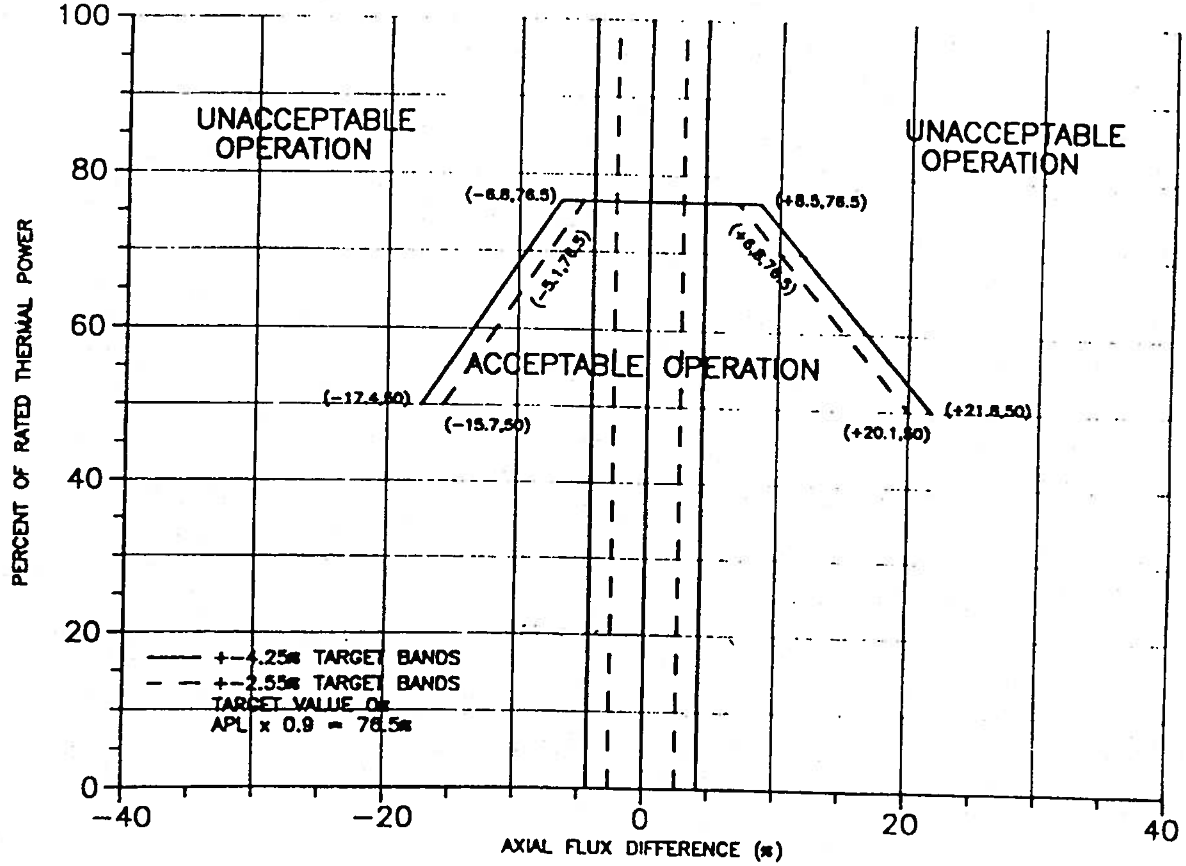
TARGET AND OPERATING BAND EXAMPLES

FIGURE 5



ERFIS CAOC WARNING/STATUS/ALARM MESSAGES

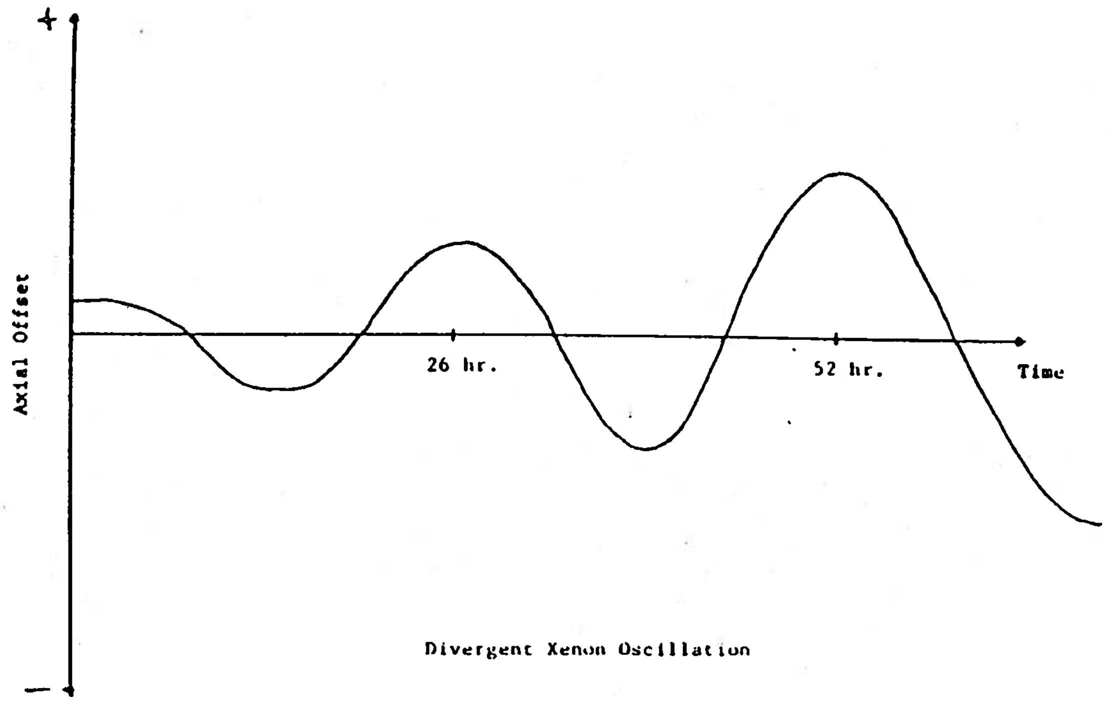
STATUS Messages - Denote a condition or message regarding a condition of interest to the Operator. These may include:
- POWER LEVEL CROSSED 15% - INCREASING - PENALTY ENABLED
- POWER LEVEL CROSSED 15% - DECREASING - PENALTY DISABLED
- POWER LEVEL CROSSED 50% - INCREASING - PENALTY RATE IS FULL
- POWER LEVEL CROSSED 50% - DECREASING - PENALTY RATE IS HALF -OPER BANDS DISABLED
- POWER LEVEL CROSSED 90%APL - INCREASING - OPER BANDS EQUAL TARGET BANDS
- POWER LEVEL CROSSED 90%APL - DECREASING - OPER BANDS RESTORED TO NORMAL
- CHANNEL #____ NOW OUT OF SERVICE
- CHANNEL #____ JUST RETURNED TO SERVICE
- CHANNEL #____ JUST CROSSED THE TARGET WARNING BAND. (decreasing)
- CHANNEL #____ JUST CROSSED THE OPER WARNING BAND. (decreasing)
- PENALTY POINT ACCUMULATION HAS BEGUN.
- PENALTY POINT ACCUMULATION HAS ENDED.

WARNING Messages - Denote a condition or message regarding a condition of impending Technical Specification violation. These may include:
- CHANNEL #____ JUST CROSSED THE TARGET WARNING BAND. (increasing)
- CHANNEL #____ JUST CROSSED THE TARGET BAND. (increasing)
- CHANNEL #__ JUST CROSSED THE OPER WARNING BAND. (increasing)
- CHANNEL #__ JUST CROSSED THE OPERATING BAND. (increasing)
- ALARM WHEN PENALTY LEVEL CROSSES 15 GOING UP.
- ALARM WHEN PENALTY LEVEL CROSSES 30 GOING UP.
- ALARM WHEN PENALTY LEVEL CROSSES 45 GOING UP.
- ALARM WHEN PENALTY LEVEL CROSSES 50 GOING UP.
- ALARM WHEN PENALTY LEVEL CROSSES 55 GOING UP.
- ALARM WHEN PENALTY LEVEL CROSSES 60 EITHER DIRECTION.

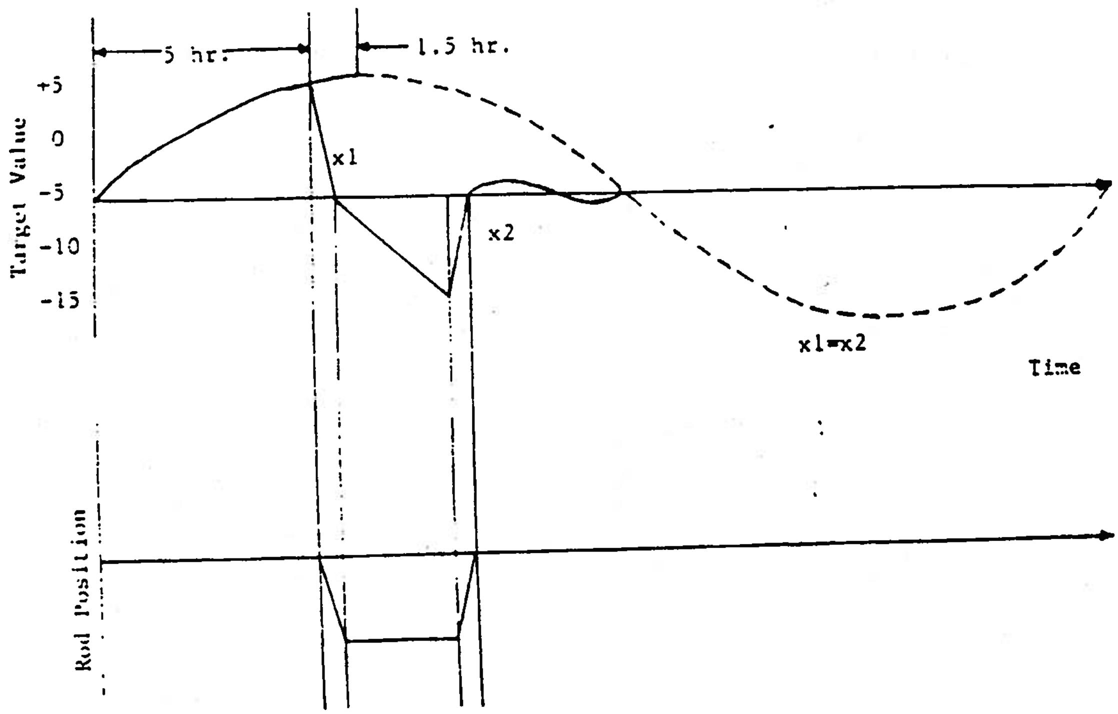
ALARM Messages - Denote a condition which violates PDC limits as set forth in plant Technical Specifications, namely:
- ALARM DUE TO 2 OR MORE CHANNELS OUTSIDE OF THE TARGET BAND WITH POWER ABOVE 90%FP OR 90%APL.
- ALARM DUE TO 2 OR MORE CHANNELS OUTSIDE OF THE OPERATING BAND WITH CORE POWER BETWEEN 50%FP AND 90%FP.
- ALARM DUE TO GREATER THAN 60 PENALTY POINTS AND CORE POWER ABOVE 50%FP.

The following special symbols are printed immediately after the AFD value for a channel, if applicable:
- **** The AFD value for the channel has just crossed a band.
- * The AFD value for the channel has not crossed a band, but is still outside of at least one.
- ?? The channel is inoperable.

DIVERGENT AXIAL OSCILLATION



DIVERGENT AXIAL OSCILLATION



NOTE: To apply numbers to the above control maneuver, assume that the Target Value is -5%, the reactor power is 75% RTP and the APL is >100% RTP. The first peak is at +5 (the dotted line representing a natural divergent oscillation). At the point the control rods are inserted, the AFD is +4%, therefore, X1=9% [+4% - (-5%)]. Stop inserting control rods at -5% and allow the AFD to go to -14% [-5% - (-9%)] with no control rod motion. When the AFD reaches -14%, pull the control rods back to their original position.

ATTACHMENT 10.5
Page 1 of 1

MANUAL AFD MONITORING LOG

This revision is the latest revision available and has been verified against the Document Management System.

______________________ (Print) ______________________ __________
Name Signature Date

Date/Time	Power %RTP	N41 AFD	N42 AFD	N43 AFD	N44 AFD	Penalty Points This Interval	Total Penalty Points	Comments	Initials

SSO Review ______________________ Date: __________

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, PDC-3, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The AFD is a measure of axial power distribution skewing to the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution and, to a lesser extent, reactor coolant temperature and boron concentrations. The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The PDC-3 axial offset control methodology (Ref. 1) entails:

a. Establishing an envelope of allowed power shapes and power densities;

b. Devising an operating strategy for the cycle that maximizes unit flexibility (maneuvering) and minimizes axial power shape changes;

c. Demonstrating that this strategy does not result in core conditions that violate the envelope of permissible core power characteristics; and

d. Demonstrating that this power distribution control scheme can be effectively supervised with excore detectors.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_Q(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also limit the range of power distributions that are assumed as initial conditions in analyzing Condition 2, 3, and 4 events. This ensures that fuel cladding integrity is maintained for the postulated accidents in Chapter 15 of the UFSAR.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement.

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator, through either the manual operation of the control banks, or automatic motion of control banks responding to

(continued)

BASES

LCO (continued)

temperature deviations resulting from either manual operation of the Chemical and Volume Control System to change boron concentration, or from power level changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 2). Separate signals are taken from the top and bottom detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detector in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as %Δ flux or %ΔI.

Part A of this LCO is modified by a Note that states the conditions necessary for declaring the AFD outside of the target band. The target bands are defined in the COLR.

With THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the AFD must be kept within the target band. With the AFD outside the target band with THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the assumptions of the accident analyses may be violated.

Parts B and C of this LCO are modified by Notes that describe how the cumulative penalty deviation time is calculated. It is intended that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. Accordingly, while THERMAL POWER is ≥ 50% RTP and < 90% RTP or 0.9 APL, whichever is less (i.e., Part B of this LCO), a 1 hour cumulative penalty deviation time limit, cumulative during the preceding 24 hours, is allowed during which the unit may be operated outside of the target band but within the acceptable operation limits provided in the COLR. This penalty time is accumulated at the rate of 1 minute for each 1 minute of operating time within the power range of Part B of this LCO (i.e., THERMAL POWER > 50% RTP but < 90% RTP or 0.9 APL, whichever is less). The cumulative penalty time is the sum of penalty times from Parts B and C of this LCO.

(continued)

BASES

LCO (continued)

Part B of the LCO is modified by a Note (Note 2) that describes the relationship of Allowable Power Level (APL) to RTP as a function of the heat flux hot channel factor at RTP, $F_Q^{RTP}(Z)$. The reactor core AFD is analyzed to 100% RTP or 100% APL, whichever is less. When $F_Q^V(Z)$ is less than its limits, 100% RTP is more limiting than 100% APL. When $F_Q^V(Z)$ is greater than its limits, 100% APL is more limiting than 100% RTP. Hence the APL results in a more restrictive operating envelope for AFD when $F_Q^V(Z)$ is greater than its limits. The K(Z) function is specified in the COLR. $F_Q^V(Z)$ is defined in the Bases of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q^V(Z)$)."

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than 50% RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Figure B 3.2.3-1 shows a typical target band and typical AFD acceptable operation limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).

(continued)

BASES

APPLICABILITY (continued)

Between 15% RTP and 100% RTP, this LCO is applicable to ensure that the distributions of xenon are consistent with safety analysis assumptions.

At or below 15% RTP and for lower operating MODES, the stored energy in the fuel and the energy being transferred to the reactor coolant are low. The value of the AFD in these conditions does not affect the consequences of the design basis events.

For surveillance of the power range channels performed according to SR 3.3.1.6, deviation outside the target band is permitted for 16 hours and no penalty deviation time is accumulated. Some deviation in the AFD is required for doing the NIS calibration with the incore detector system. This calibration is performed every 92 days.

Low signal levels in the excore channels may preclude obtaining valid AFD signals below 15% RTP.

ACTIONS

A.1

With the AFD outside the target band and THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less, the assumptions used in the accident analyses may be violated with respect to the maximum heat generation. Therefore, a Completion Time of 15 minutes is allowed to restore the AFD to within the target band because xenon distributions change little in this relatively short time.

B.1

If the AFD cannot be restored within the target band, then reducing THERMAL POWER to < 90% RTP or 0.9 APL, whichever is less places the core in a condition that has been analyzed and found to be acceptable, provided that the AFD is within the acceptable operation limits provided in the COLR.

The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP or 0.9 APL whichever is less without allowing the plant to remain in an unanalyzed condition for an extended period of time.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With THERMAL POWER < 90% RTP or 0.9 APL, whichever is less but ≥ 50% RTP, operation with the AFD outside the target band is allowed for up to 1 hour if the AFD is within the acceptable operation limits provided in the COLR. With the AFD within these limits, the resulting axial power distribution is acceptable as an initial condition for accident analyses assuming the then existing xenon distributions. The 1 hour cumulative penalty deviation time restricts the extent of xenon redistribution. Without this limitation, unanalyzed xenon axial distributions may result from a different pattern of ~~xenon~~ buildup and decay. The reduction to a power level < 50% RTP puts the reactor at a THERMAL POWER level at which the AFD is not a significant accident analysis parameter. Restoration of cumulative penalty time to less than one (1) hour prior to increasing THERMAL POWER to above ≥ 50% RTP in accordance with Required Action C.2 ensures that the AFD will be within the core analysis.

If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. (Any AFD within the target band is acceptable regardless of its relationship to the acceptable operation limits.) The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power.

Condition C is modified by a Note that requires that Required Actions C.1 and C.2 must be completed whenever this Condition is entered.

D.1

If Required Action C.1 is not completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid.

(continued)

BASES

ACTIONS

D.1 (continued)

Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

This Required Action must also be implemented either if the cumulative penalty deviation time is > 1 hour during the previous 24 hours, or the AFD is not within the target band and not within the acceptable operation limits.

Condition D is modified by a Note that requires Action D.1 be completed whenever this Condition is entered.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AFD is monitored on an automatic basis using the unit process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFDs for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is > 90% RTP or 0.9 APL, whichever is less. During operation at THERMAL POWER levels < 90% RTP or 0.9 APL, whichever is less but > 15% RTP, the computer sends an alarm message when the cumulative penalty deviation time is > 1 hour in the previous 24 hours.

This Surveillance verifies that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Furthermore, any deviations of the AFD from the target band that is not alarmed should be readily noticed.

SR 3.2.3.2

With the AFD monitor alarm inoperable, the AFD is monitored to detect operation outside of the target band and to compute the penalty deviation time. During operation at ≥ 90% RTP or 0.9 APL, whichever is less, the AFD is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.2 (continued)

monitored at a Surveillance Frequency of 15 minutes to ensure that the AFD is within its limits at high THERMAL POWER levels. At power levels < 90% RTP or 0.9 APL, whichever is less, but > 15% RTP, the Surveillance Frequency is reduced to 1 hour because the AFD may deviate from the target band for up to 1 hour using the methodology of Parts B and C of this LCO to calculate the cumulative penalty deviation time before corrective action is required.

SR 3.2.3.2 is modified by a Note that states that monitored and logged values of the AFD are assumed to exist for the preceding 24 hour interval in order for the operator to compute the cumulative penalty deviation time. The AFD should be monitored and logged more frequently in periods of operation for which the power level or control bank positions are changing to allow corrective measures when the AFD is more likely to move outside the target band.

SR 3.2.3.3

Measurement of the target flux difference is accomplished by taking a flux map when the core is at equilibrium xenon conditions, preferably at high power levels with the control banks nearly withdrawn. This flux map provides the equilibrium xenon axial power distribution from which the target value can be determined. The target flux difference varies slowly with core burnup.

A Frequency of 31 EFPD after each refueling and 31 EFPD thereafter for remeasuring the target flux differences adjusts the target flux difference for each excore channel to the value measured at steady state conditions.

A Note modifies this SR to allow the predicted beginning of cycle AFD from the cycle nuclear design to be used to determine the initial target flux difference after each refueling.

A second Note modifies this SR to require that the target flux difference be determined in conjunction with the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with SR 3.2.1.1. This is a requirement of the PDC-3 Axial Offset Control Methodology.

(continued)

BASES (continued)

REFERENCES

1. ANF-88-054 (Proprietary). "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352, July 1988 (Submitted to NRC by CP&L letter dated August 24, 1989).

2. UFSAR Section 7.2.1.1

3. XN-NF-77-57(P)(A) (Proprietary). "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II," Supplement 2 and Supplement 2, Addendum 1," Exxon Nuclear Company, Richland WA 99352, October 1982, page 34.

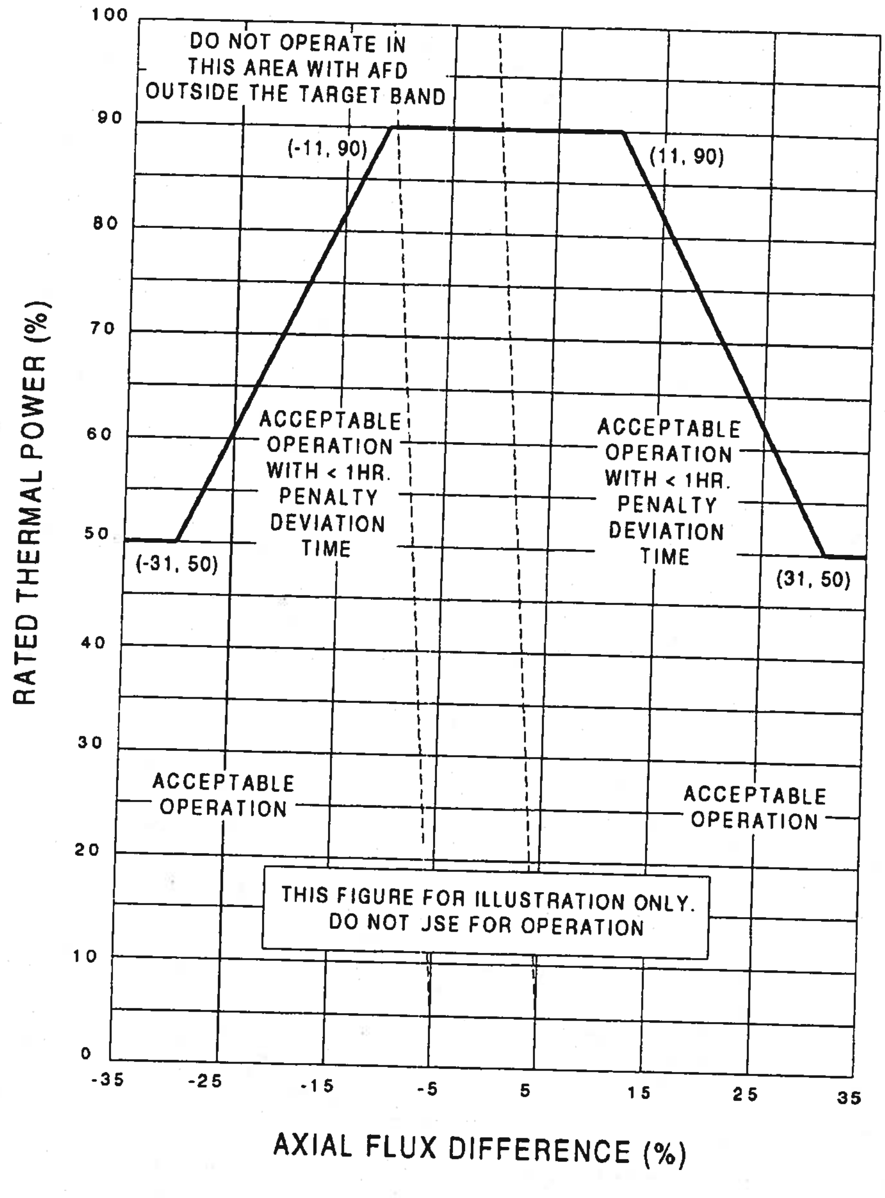


Figure B 3.2.3A-1 (Page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
and Target Band Limits as a Function
of RATED THERMAL POWER

93. 029 A2.01 SRO 001

Given the following plant conditions:

- The plant is at 100% RTP.
- Maintenance is preparing to enter the CV to repair a leak on a sensing line.
- CV Pressure Relief is in progress with CV pressure currently at 0.05 psig.
- CV Purge Gaseous Waste Release Permit has been issued to allow the CV Purge to commence.
- Both R-11 AND R-12 are in service.

Which ONE (1) of the following completes the statements below?

The CV Purge and CV Pressure Relief __(1)__ be performed concurrently.

R-14C, Auxiliary Building Vent Stack Noble Gas Monitor, __(2)__ required to be operable during the CV Purge.

A. (1) may

(2) is NOT

B. (1) may

(2) is

C.✓ (1) may NOT

(2) is NOT

D. (1) may NOT

(2) is

The correct answer is C.

A. Incorrect. A CV Purge and CV Pressure Relief may not be performed concurrently. This is 3.6.3 prohibits opening the 42 inch purge and 6 inch pressure relief valves simultaneously. R-14C is normally in service during a pressure relief and purge, but is not required as long as R-11 AND R-12 are both operable. This requirement is specified in the ODCM and EMP-022, Gaseous Waste Release Permits.

B. Incorrect. See discussion in "A" above.

C. Correct.

D. Incorrect. The first part of the distractor is correct. See discussion in "A" above.

Question 93
Tier/Group 2 / 2
K/A Importance Rating - RO 2.6 SRO 2.8

Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Maintenance or other activity taking place inside containment

Reference(s) - Sim/Plant design, EMP-022, ODCM, OP-921
Proposed References to be provided to applicants during examination - None
Learning Objective - CVHVAC 007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.5 / 43.5 / 45.3 / 45.13
Comments -

SRO: Knowledge Tech Spec requirements that are below the double line and greater than 1 hour. Application of Offsite Dose Calculation Manual (ODCM) requirements. Knowledge of requirements for gaseous release approvals, i.e., release permits.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met. <u>OR</u> 42 inch penetration (Supply or Exhaust) purge valves open and 6 inch penetration (pressure or vacuum relief) valves open simultaneously.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	-------------------NOTE------------------- The 42 inch and 6 inch valves may not be open simultaneously. -- Verify each 42 inch purge supply and exhaust valve and each 6 inch pressure and vacuum relief valve is closed, except when the valves are open for safety related reasons, or for tests or Surveillances that require the valves to be open.	31 days

(continued)

GASEOUS WASTE RELEASE PERMIT - BATCH CONTAINMENT

RELEASE NUMBER: ____________________ SSN: ________________ DATE: ________________

This revision is the latest revision available as verified by:

______________________ ________ ____________________________ ________
Name (Print) Initial Signature Date

PART I: PRE-RELEASE INFORMATION (E&C)			
Pressure Relief ________ Batch Purge ________ CV PR Via Vacuum Relief ________ Other ________ (check one)	Estimated Release Start ________ Date ________ Time Estimated Release Stop ________ Date ________ Time		
R-11/12 Sample Point during release: C.V. or PV (Circle One)			
Monitor	Setpoint	Basis (Circle One)	
R-11	CPM	EC Activity	
R-12	CPM	EC Activity	
R-14C (3)	CPM	EC Activity	NA

This release can be made within the limits of 10CFR20 and 10CFR50 using the setpoints and restrictions stated herein.

Prepared By: ______________________ Peer Review: ______________________

E&C Supervisor: ______________________

PART II: RADIATION MONITOR INFORMATION (OPS and E&C)			(CR 98-00002)
READING	R-11	R-12	R-14C (3)
PRIOR (4)	CPM	CPM	CPM
SOURCE CHECK (5)	OPS INI.	OPS INI.	
SETPOINT VERF. AT (6)	CPM	CPM	CPM
UPDATE STATUS BOARD (7)			OPS INI.
DURING RELEASE	CPM	CPM	CPM
AFTER RELEASE	CPM	CPM	CPM
SETPOINT RETURNED TO (8)	CPM	CPM	CPM
STATUS BOARD UPDATED (7)			OPS INI.

Approved for Release: ______________________________ (CR 97-00059)
Shift Manager

GASEOUS WASTE RELEASE PERMIT - BATCH CONTAINMENT

Release # ____________________

PART III: RELEASE INFORMATION[1] (OPS)			
Ensure at least one Aux. Building fan running. ___________ (OPS) Ensure no WGDT release in progress. ___________ (OPS)			
RELEASE	DATE	TIME	PSIG
START PRESSURE RELIEF OR PURGE FAN			
STOP (2)			
			Δ P

Notes:

1. N/A all blanks not applicable.
2. For a Batch C.V. Purge becoming a Continuous C.V. Purge, the stop time is the start time of the Continuous C.V. Purge.
3. E&C Technician will perform setpoint changes for R-14C as per EMP-013. R-14C setpoint is not required for a CV PR via containment vacuum relief, NA these blanks.
4. If Rad Monitor is out-of-service, refer to Section 7.3.3 **OR** 7.3.4 of EMP-022. (ITS LCO 3.3.6)
5. Source check for R-12 required prior to each C.V. Release (ODCM Table 3.11-1, item 2.b). If R-12 fails source check, refer to Section 7.3.3 **OR** 7.3.4. (ITS LCO 3.3.6)
6. If current R-11/R-12 or R-14C setpoints are more conservative than those in PART I, the setpoint need not be changed. Log cpm that monitor setpoint was verified at.
7. Verify status board in control room has correct setpoint.
8. Upon completion of C.V. Batch Releases, notify E&C to return R-14C setpoint to listed value on current weekly Continuous Plant Vent release permit (not required for CV PR via Containment Vacuum Relief) and reset R-11 and R-12; if changed, as per OMM-014.

Completed by: ____ (E&C Technician)

Completed by: ____ (Control Operator)

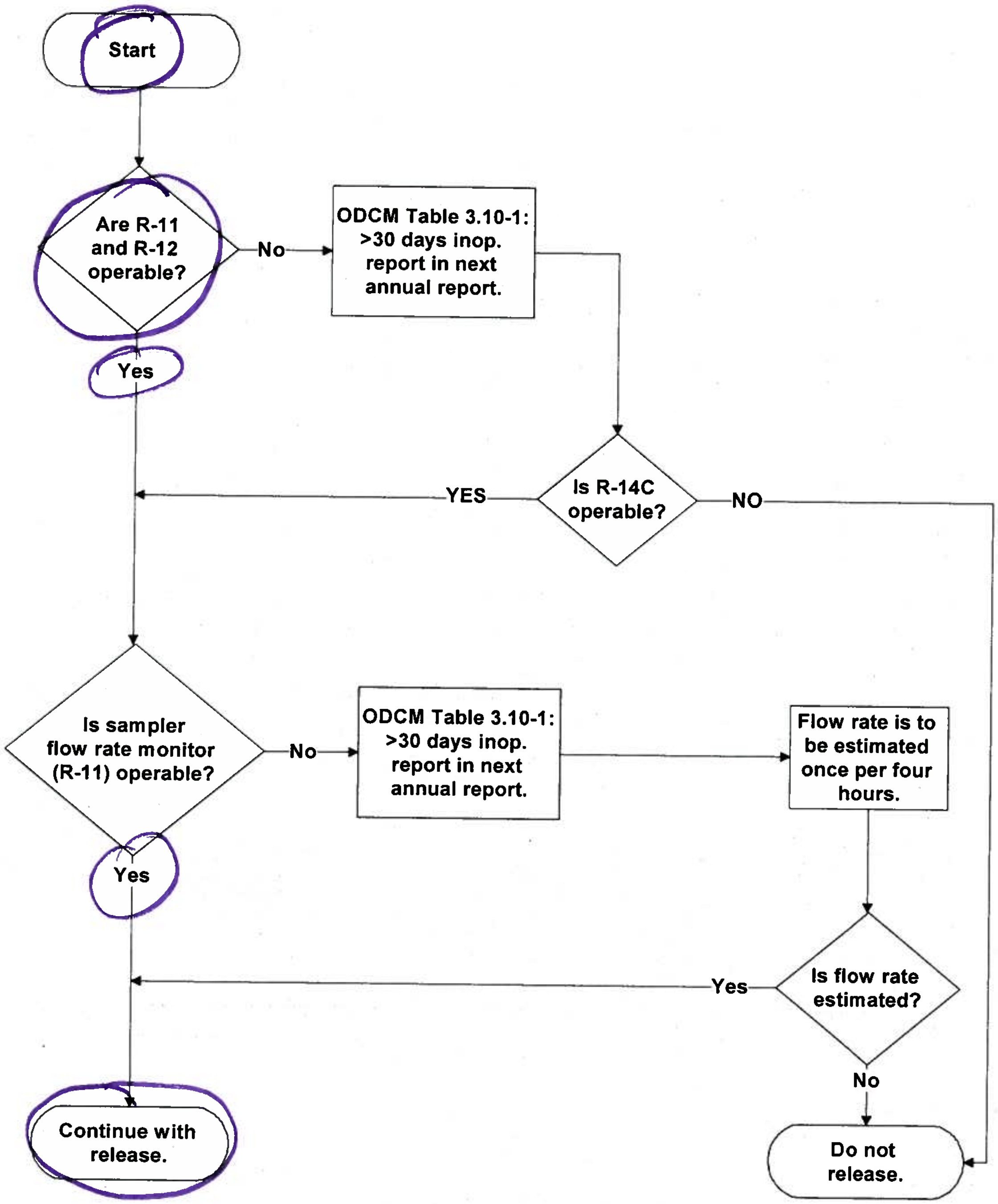
Reviewed by: ____ (Shift Manager)

POST RELEASE REVIEW

Release Posted By: ________________________________ Date: ____________

______________________________ ____________
Reviewed By Date

7.3.3 Minimum Instrumentation Requirements for Containment Vessel Releases Via Plant Vent - Modes 1, 2, 3, 4 **AND** movement of recently irradiated fuel.



R-14C

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (Continued)		
c. Radioiodine Sampler	1	With the number of channels operable less than the MCO requirements: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that a continuous sample is collected utilizing auxiliary sampling equipment as required by Table 3.12-1. (note 1)
d. Particulate Sampler	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may be continued, provided that a continuous sample is collected utilizing auxiliary sampling equipment as required by Table 3.12-1. (note 1)
e. Sampler flow rate monitor	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided the flow rate is estimated once per 4 hours.

12/06

* MCO - Minimum Channels Operable

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (Continued)		
f. Plant Vent flow rate	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that flow rate is estimated once per 4 hours.
2. Containment Vessel via Plant Vent		
R-12 a. Radionoble gas monitor (R-12) provides automatic termination of Containment Vessel releases upon exceeding alarm/trip Setpoint.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)
R-11 b. Radioparticulate Monitor (R-11) provides automatic termination of containment vessel releases exceeding alarm/trip setpoints.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)

* MCO - Minimum Channels Operable

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
2. Containment Vessel via Plant Vent (Continued)		
c. Sampler flow rate monitor (R-11)	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that either the Plant Vent Radionoble Gas Monitor (R-14C) is operable or the flow rate is estimated once per 4 hours. (note 2)
3. Fuel Handling Building Lower Level Exhaust Vent		
a. Radionoble gas monitor (R-20)	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that grab samples are taken once per 12 hours and analyzed for radionoble gases within 24 hours.
b. Sampler flow rate monitor (R-20)	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided the flow rate is estimated once per 4 hours.

* MCO - Minimum Channels Operable

05/10

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
6. Radwaste Building Exhaust (Continued)		
c. Sampler flow rate gauge	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided the flow rate is estimated once per 4 hours.
7. Deleted.	NA	NA

* MCO - Minimum Channels Operable

12/06

NOTES TO TABLE 3.10-1

Note 1 - No auxiliary sampling is required for periods when normal sampling is off $\leq$ 45 minutes.

Note 2 - This MCO is required during Modes 1, 2, 3, 4, and during the movement of recently irradiated fuel assemblies within the containment.

8.4.1.1 (Continued)

INIT

> **NOTE:** In MODES 1, 2, 3, 4, **AND** when moving recently irradiated fuel in the Containment, the Containment Ventilation Isolation signal from R-11 **AND** R-12 is required to be OPERABLE. (ITS 3.3.6)
>
> In accordance with ODCM requirements, one of the following is required for monitoring during Containment Purging (ODCM Table 3.10-1):
>
> R-11 **AND** R-12
>
> **OR**
>
> R-14C

e. **IF** R-11 is in service, **THEN PERFORM** the following:

1) **IF** the current R-11 setpoint is higher than the setpoint on the release permit **OR** it is desired to adjust the RMS setpoint, **THEN ADJUST** RMS setpoint IAW the new value on the release permit. ____

2) **PERFORM** a CHANNEL CHECK on R-11. (ITS SR 3.3.6.1 TBL 3.3.6-1 Item 3) ____

f. **IF** R-12 is in service, **THEN PERFORM** the following:

1) **IF** the current R-12 setpoint is higher than the setpoint on the release permit **OR** it is desired to adjust the RMS setpoint, **THEN ADJUST** RMS setpoint IAW the new value on the release permit. ____

2) **PERFORM** a CHANNEL CHECK on R-12. (ITS SR 3.3.6.1 TBL 3.3.6-1 Item 3) ____

94. G2.1.31 SRO 001

Given the following plant conditions:

- The Motor Driven Fire Pump (MDFP) was started remotely per OST-603, MDFP and EDFP Test (Weekly).
- The MDFP supply breaker tripped open during OST-603.

Which ONE(1) of the following completes the statement below?

MDFP indication __(1)__ available on the Containment Fire Protection Panel.

The Fire Suppression Water Supply System is considered __(2)__ IAW FP-012, Fire Protection Systems Minimum Equipment and Compensatory Actions.

A. (1) is NOT

(2) Operable

B.✓ (1) is

(2) Operable

C. (1) is NOT

(2) Inoperable

D. (1) is

(2) Inoperable

The correct answer is B.

A. Incorrect. The pump status lights on the CFPP in the control room will show dual indication. Second half of the distractor correct.

B. Correct.

C. Incorrect. The pump status lights on the CFPP in the control room will show dual indication. Per FP-012, only ONE fire pump is required for the Fire Suppression Water Supply System to be considered OPERABLE. This requirement is met by the Engine Driven Fire Pump being operable. An EIR will be filled out on the MDFP with a seven day return to service time or initiate an NCR.

D. Incorrect. First part of the distractor is correct. Per FP-012, only ONE fire pump is required for the Fire Suppression Water Supply System to be considered OPERABLE. This requirement is met by the Engine Driven Fire Pump being operable. An EIR will be filled out on the MDFP with a seven day return to service time or initiate an NCR.

Exert from FP-012.

8.2 Fire Suppression Water Supply System

NOTE: *Each fire pump provides 100% of the required flow.* ***Therefore, the loss of one (1) fire pump does not cause the Fire Suppression Water Supply System to be inoperable*** *as described in 8.2.3.*

8.2.1 The Fire Suppression Water Supply System shall be operable with:

1. *Two (2) fire pumps, each with a capacity of 2,500 gpm, with their discharge aligned to the yard loop,* ***AND***

2. *An operable flow path capable of taking suction from the Unit 2 intake structure* ***AND*** *transferring the water through distribution piping with operable sectionalizing* ***OR*** *isolation valves to the systems identified in Sections 8.3 and 8.4.*

8.2.2 With less than the above required equipment operable, restore the inoperable equipment to operable status within seven (7) days.

Question 94
Tier 3
K/A Importance Rating - RO 4.2 SRO 4.2

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Reference(s) - Sim/Plant design, FP-012, System Description
Proposed References to be provided to applicants during examination - None
Learning Objective - FPW007
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10, 45.12
Comments - K/A match because candidate must know where the status of the MDFP can be obtained in the control room.

SRO: This question is SRO level because it is administration of fire protection program requirements for determining operability of the Fire Suppression Water Supply System.

Reviewed and approved by MAB.

95. G2.1.41 SRO 001

Given the following conditions:

- The plant is in MODE 6 for Refueling.
- Core offload is in progress with a Fuel Assembly on the Manipulator.
- Refueling Cavity level is lowering.

Which ONE (1) of the following completes the statements below?

The Fuel Assembly from the Manipulator is required to be placed in the core, in __(1)__ as required by __(2)__ .

A. (1) in its original location

(2) AOP-013, Fuel Handling Accident

B. (1) any location that is bordered by 2 other assemblies

(2) AOP-013, Fuel Handling Accident

C.✓ (1) in its original location

(2) AOP-020, Loss of Residual Heat Removal (Shutdown Cooling)

D. (1) in any location that is bordered by 2 other assemblies

(2) AOP-020, Loss of Residual Heat Removal (Shutdown Cooling)

REVISE

The correct answer is C.

A: Incorrect - Place the fuel assembly back in a location where subcritical configuration was known. The transfer cart must be on the CV side to be able to shut the Gate Valve.

B: Incorrect - Placing a fuel assembly in an unanalyzed core position could result in a loss or reduction of the required shutdown margin. The transfer cart must be on the CV side to be able to shut the Gate Valve.

C: Correct - Place the fuel assembly back in a location where subcritical configuration was known. The transfer cart must be on the CV side to be able to shut the Gate Valve.

D: Incorrect - Placing a fuel assembly in an unanalyzed core position could result in a loss or reduction of the required shutdown margin. The transfer cart must be on the CV side to be able to shut the Gate Valve.

Question 95
Tier 3
K/A Importance Rating - RO 4.6 SRO 4.3

Knowledge of the refueling processes.

Reference(s) - AOP-020, Section B., AOP-013
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-020-04
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.2 / 41.1 / 43.6 / 45.13
Comments -

SRO: Question involves assessing plant facility conditions and having knowledge of the content of the mitigating procedure. Fuel handling in the CV is an SRO only function. Refuel floor SRO responsibilities.

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

Section B

Loss Of RHR Inventory - Vessel Head Off

(Page 2 of 10)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
3.	Notify Refueling Personnel To Perform The Following: a. Place any fuel assembly in transit in one of the following locations: • Original Core location • Upender • Storage location approved by FMP-019, Fuel and Insert Shuffle b. Place any Reactor Vessel Upper or Lower Internals in transit in one of the following locations: • Reactor Vessel (preferred location) • Designated storage location in transfer canal	
4.	Notify Refueling Personnel to Perform the Following: a. Verify Fuel Transfer Conveyer Car Location - IN CONTAINMENT b. Verify CV Upender Position - HORIZONTAL c. Verify CLOSED The SFP GATE VALVE	
5.	Check APP-036-B6, Spent Fuel Pit Lo Level - EXTINGUISHED	Refer to AOP-036, SFP Events.
6.	Check Cavity Seal OR Sand Plug Failure - IN PROGRESS	Observe the NOTE prior to Step 15 and Go To Step 15.

CONTINUOUS USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL
VOLUME 3
PART 5
ABNORMAL OPERATING PROCEDURE

AOP-013

FUEL HANDLING ACCIDENT

REVISION 13

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides the instructions for actions to be taken for a fuel handling accident in Containment, the Spent Fuel Pit Area or the New Fuel Storage Area.

NOTE

- For the purposes of this procedure an "Accident" is a drop of a fuel assembly OR other event that has the potential to cause damage to the assembly.
- Minor bumps of assemblies being moved AND leaning assemblies are NOT considered accidents for this procedure entry.

2. ENTRY CONDITIONS

Upon receiving a report that a fuel handling accident, as described above, has occurred.

- END -

96. G2.2.35 SRO 001

Given the following plant conditions:

- A plant heatup from a refueling outage is currently in progress.
- Highest available RCS temperature is 325°F and rising.
- The "A" MDAFW Pump has been declared OOS.

Which ONE (1) of the following completes the statement below?

The current technical specification operational MODE is __(1)__ and IAW the technical specifications a change to the next higher MODE based on conditions given is __(2)__.

A. (1) 3

(2) allowed as long as a risk assessment addressing the inoperable MDAFW Pump is performed

B. (1) 4

(2) allowed as long as a risk assessment addressing the inoperable MDAFW Pump is performed

C. (1) 3

(2) NOT allowed

D✓ (1) 4

(2) NOT allowed

The correct answer is D.

A. Incorrect. LCO 3.0.4b (risk assessment evaluation) is not applicable for ITS 3.7.4, Auxiliary Feedwater (AFW) System. Mode is incorrect.

B. Incorrect. LCO 3.0.4b (risk assessment evaluation) is not applicable for ITS 3.7.4, Auxiliary Feedwater (AFW) System. Mode is correct.

C. Incorrect. Mode is incorrect. Mode 3 is not entered until temperature is greater than or equal to 350°F.

D. Correct. LCO 3.0.4b (risk assessment evaluation) is not applicable for ITS 3.7.4, Auxiliary Feedwater (AFW) System.

Exert from LCO 3.0.4

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time, or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Question 96
Tier 3
K/A Importance Rating - RO 3.6 SRO 4.5

Ability to determine Technical Specification Mode of Operation.

Reference(s) - Sim/Plant design, ITS Section 1.1, 3.0, 3.7.4
Proposed References to be provided to applicants during examination - None
Learning Objective - Intro to Tech Specs 001, 005; AFW 012
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 41.10 / 43.2 / 45.13
Comments -

SRO: Application of generic Limiting Condition of Operation requirements (LCO 3.0.1 thru 3.0.7)

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and 3.0.7.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time, or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)	the loss of safety function exists are required to be entered. When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

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

-----NOTE-----
LCO 3.0.4.b is not applicable.

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)

ACTIONS (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

97. G2.2.42 SRO 001

Given the following plant conditions:

- The plant is at 100% RTP.
- Chemistry reports the following results from a primary to secondary leakage rate determination:
  - "A" S/G leakage is 0.04 gpm
  - "B" S/G leakage is 0.05 gpm
  - "C" S/G leakage is 0.06 gpm

Which ONE (1) of the following completes the statement below?

Technical Specification 3.4.13, RCS Operational Leakage, is required to be entered due to __(1)__ S/G primary to secondary LEAKAGE being exceeded.

Entry into Mode 3 is required within __(2)__ hours.

A. (1) total

(2) 6

B.✔ (1) any one

(2) 6

C. (1) total

(2) 36

D. (1) any one

(2) 36

The correct answer is B.

A. Incorrect - Although all three S/Gs have tube leakage, only "C" S/G is greater than the ITS limit of 75 gpd. "A" S/G is 43.2 gpd and "B" S/G is 57.6 gpd. "C" S/G has primary to secondary leakage of 100.8 gpd. Second half of the answer is correct.

B. Correct.

C. Incorrect - Although all three S/Gs have tube leakage, only "C" S/G is greater than the ITS limit of 75 gpd. "A" S/G is 43.2 gpd and "B" S/G is 57.6 gpd. "C" S/G has primary to secondary leakage of 100.8 gpd. The second half of the distractor is valid for primary to secondary leakage greater than 75 gpd but less than 100 gpd.

D. Incorrect. The first part of the distractor is correct. The second half of the distractor is valid for primary to secondary leakage greater than 75 gpd but less than 100 gpd.

Question 97
Tier 3
K/A Importance Rating - RO 3.9 SRO 4.6

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Reference(s) - Sim/Plant design, ITS 3.4.13, AOP-035
Proposed References to be provided to applicants during examination - None
Learning Objective - Intro to Tech Specs 005, RCS 013
Question Source - NEW
Question Cognitive Level - H
10 CFR Part 55 Content - 41.7 / 41.10 / 43.2 / 43.3 / 45.3
Comments -

SRO: Application of required actions in ITS.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. ~~10 gpm identified~~ LEAKAGE; ~~and~~

d. 75 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. **OR** Pressure boundary LEAKAGE exists. **OR** Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. **AND** B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-------------------NOTES------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	-------------------NOTE------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 75 gallons per day through any one SG.	72 hours

98. G2.3.4 SRO 001

Given the following plant conditions:

- A manual reactor trip and safety injection were initiated from 100% RTP.
- The crew has transitioned to PATH-2 due to a tube rupture on "B" S/G.
- MS-V1-8B, SDAFW Steam Shutoff Valve, MOV breaker tripped.
- R-31B, Steamline Radiation Monitor, has indicated approximately **250 mrem/hr** throughout the event.
- The OAO has been directed by the **CRS** to locally close MS-V1-8B to isolate the potential radioactive release path.
- The OAO currently has an accumulated annual dose of 1000 mrem TEDE from Progress Energy and 500 mrem TEDE while employed by another utility.

Which ONE(1) of the following completes the statement below?

Based on **DOS-NGGC-0004, Administrative Dose Limits**, __(1)__ is the **maximum** dose that the operator can receive to isolate this release path and based on the conditions given a(n) __(2)__ should be declared.

(REFERENCE PROVIDED)

A.✓ (1) 1000 mrem

(2) Site Area Emergency

B. (1) 500 mrem

(2) Site Area Emergency

C. (1) 1000 mrem

(2) Alert

D. (1) 500 mrem

(2) Alert

The correct answer is A.

A. Correct - The dose given would place the worker at the 2000 mrem Progress Energy Annual Administrative limit. The SAE is correct.

B. Incorrect - The dose given would place the worker at 2000 mrem dose from all sources. The Progress Energy Annual Administrative limit is 2000 mrem from Progress Energy not to exceed 4000 mrem total.

C. Incorrect. The dose given would place the worker at the 2000 mrem Progress Energy Annual Administrative limit. If the candidate only evaluated the Fission Product Barrier Matrix he/she would determine that only the RCS Barrier is lost and this would meet the criteria for an Alert declaration. However, R-31B indication of 250 mrem/hr exceeds the SAE limit of 220 mr/hr as specified in Table R-1, Effluent Monitor Classification Thresholds.

D. Incorrect - The dose given would place the worker at 2000 mrem dose from all sources. The Progress Energy Annual Administrative limit is 2000 mrem from Progress Energy not to exceed 4000 mrem total. If the candidate only evaluated the Fission Product Barrier Matrix he/she would determine that only the RCS Barrier is lost and this would meet the criteria for an Alert declaration. However, R-31B indication of 250 mrem/hr exceeds the SAE limit of 220 mr/hr as specified in Table R-1, Effluent Monitor Classification Thresholds.

Question 98
Tier 3
K/A Importance Rating - RO 3.2 SRO 3.7

Knowledge of radiation exposure limits under normal or emergency conditions.

Reference(s) - Sim/Plant design, EALs, DOS-NGGC-0004
Proposed References to be provided to applicants during examination - EAL Matrix
Learning Objective - EAL 004
Question Source - BANK
Question Cognitive Level - H
10 CFR Part 55 Content - 41.12 / 43.4 / 45.10
Comments -

SRO: Emergency Action Level Classification and determination of administrative dose limits.

x Progress Energy

R
Reference
Use

NUCLEAR GENERATION GROUP

STANDARD PROCEDURE

Volume 99

Book/Part 99

DOS-NGGC-0004

ADMINISTRATIVE DOSE LIMITS

REVISION 12

NGG
Think Fleet

5.0 PREREQUISITES

N/A

6.0 PRECAUTIONS AND LIMITATIONS

N/A

7.0 SPECIAL TOOLS AND EQUIPMENT

N/A

8.0 ACCEPTANCE CRITERIA

N/A

9.0 INSTRUCTIONS

R2.1 9.1 Adult Occupational Dose Limits

9.1.1 Whole Body - The more limiting of a total effective dose equivalent equal to 5 rem or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye equal to 50 rem.

9.1.2 Skin - A shallow dose equivalent equal to 50 rem.

9.1.3 Lens of Eye - A lens dose equivalent equal to 15 rem.

9.1.4 Extremities - A shallow dose equivalent equal to 50 rem.

9.2 Occupational Dose to Minors

Minors shall not be employed to work in radiation control areas, although they may enter as visitors.

9.3 Progress Energy Annual Administrative Dose Limits

9.3.1 0.5 rem Progress Energy dose if non-Progress Energy dose for the current year has not been determined. No dose extension is permitted.

9.3.2 2 rem Progress Energy dose not to exceed 4 rem total dose if non-Progress Energy dose for the current year has been determined.

Category: R – Abnormal Rad Release / Rad Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release

EAL:

RS1.1 Site Area Emergency

Valid reading on **any** radiation monitors that exceeds or is expected to exceed Table R-1 column "SAE" for ≥ 15 minutes (Note 1)

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should **not** be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification is warranted, should be subsequently escalated, or if protective actions should be revised.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Plant Vent	R-14C	---------	---------	---------	2 X alarm
		R-14D	2.3E5 cpm	2.3E4 cpm	2.3E3 cpm	---------
		R-14E	1.5E3 cpm	1.5E2 cpm	1.5E1 cpm	---------
	FHB Exhaust	R-20	---------	---------	---------	2 X alarm
	FHB Exhaust HR	R-30	1.0E4 mR/hr	1.0E3 mR/hr	1.0E2 mR/hr	---------
	Main Steamline	R-31 A/B/C	2.2E3 mR/hr	2.2E2 mR/hr	2.2E1 mR/hr	---------
Liquid	Liquid Waste Disposal	R-18	--------	--------	200 X alarm*	2 X alarm*
	SGBD Effluent	R-19A/B/C	--------	--------	200 X alarm*	2 X alarm*
	Condensate Polisher	R-37	--------	--------	200 X alarm*	2 X alarm*

* With effluent discharge **not** isolated

Mode Applicability:

All

99. G2.4.30 SRO 001

Given the following plant conditions:

- The plant was operating at 100% RTP when "C" RCP trips.
- All AFW pumps received an Auto-start signal.

Which one of the following completes the statement below?

IAW AP-030, NRC Reporting Requirements, the NRC is required to be notified within __(1)__ hour(s) due to the automatic RPS initiation and within __(2)__ hours for the ESFAS actuation.

A. (1) 1

(2) 4

B. (1) 1

(2) 8

C. (1) 4

(2) 4

D.✓ (1) 4

(2) 8

The correct answer is D.

A. Incorrect. Candidate may think that the NRC must be notified via AP-030 due to a reactor trip and four hours for the ESFAS notification.

B. Incorrect. Candidate may think that the NRC must be notified via AP-030 due to a reactor trip. The second half of distractor is correct.

C. Incorrect. Candidate may think that both actions require a four hour notification.

C. Correct.

Question 99
Tier 3
K/A Importance Rating - RO 2.7 SRO 4.1

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Reference(s) - AP-030
Proposed References to be provided to applicants during examination - None
Learning Objective - AOP-030-003
Question Source - New
Question Cognitive Level - H
10 CFR Part 55 Content - 41.10 / 43.5 / 45.11
Comments -

SRO: Off-site reporting requirements.

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

Distractor

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, four-hours, or eight hours to the NRC. ETS Telephones, which are identified, are located in the Control Room, the TSC, and the EOF. In the event that the ETS is not available, 10 CFR 50.72(a)(2) permits the use of commercial telephone.

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) 10 CFR 72.75(a)	Emergency Unusual Event Alert Site Area Emergency General Emergency ISFSI	HBRSEP shall notify the NRC of the declaration of any of the Emergency Classes specified in the Emergency Plan. (See EPNOT-01)	– 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.
DEVIATION FROM TS (10 CFR 50.54(X)) 10 CFR 50.72(b)(1)	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).

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) UFSAR Section 17.3A, Paragraph 3.1.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. NRC Region II must also be notified within 1 hour and the Vice President - Robinson Nuclear Plant within 24 hours	– The limits of TS Figure 2.1.1-1 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 (SEC-NGGC-2147)

* In response to NRC Bulletin 2005-02, RNP committed to make an accelerated call to the NRC within approximately 15 minutes following discovery of an imminent threat or attack against the station. The primary purpose is to allow for the NRC to timely notify other licensees of a potential common threat. The accelerated call should not be allowed to interfere with plant or personnel safety, physical security response, or notification of local law enforcement agencies. The information provided in the accelerated call can be limited to:

- Site name
- Emergency Classification – if already determined – do not delay call for the purpose of classifying
- Nature of the threat – briefly described, if known, including the type of attack (e.g., armed assault by land, water or aircraft) and the attack status (e.g., imminent, in progress, or repelled)

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 (SEC-NGGC-2147)
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 (SEC-NGGC-2147)
FAILURE, DEGRADATION, OR DISCOVERED VULNERABILITY OF SAFEGUARD SYSTEM 10 CFR 73, Appendix G, I(c)	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.	- Procedure SEC-NGGC-2147
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.

* See footnote on the previous page regarding a goal for a 15 minute call to the NRC in regard to an imminent security threat or attack.

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
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 20.2202(a)(2)	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 - 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 immediately notify the NRC Operations Center via ETS, when:

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 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 (SEC-NGGC-2147)
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 or paragraphs (a), (b), (c), or (d) of 10 CFR 72.75, 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) 10 CFR 72.75(f)(1)	Degradation Emergency Class Change Update Termination ISFSI	(i) any further degradation in the level of safety of the plant or ISFSI or other worsening 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 EPNOT-01
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(2) 10 CFR 72.75(f)(2)	Result Evaluation Effectiveness Unknown ISFSI	(i) the results of ensuing evaluations or assessments of plant or ISFSI conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant or ISFSI behavior that is not understood.	
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(3) 10 CFR 50.72.75(f)(3)	Open Continuous Communication ISFSI	Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.	– Refer to EPNOT-01

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 26.77(c)	Alcohol Influence Substance NRC employee FFD Fitness for Duty	If HBRSEP has a reasonable belief that an NRC employee or NRC contractor may be under the influence of any substance, or is otherwise unfit for duty, the licensee or other entity may not deny access but shall escort the individual. In any such instance, the licensee or other entity shall immediately notify the Region II Administrator by telephone, followed by written notification (e.g., e-mail or fax) to document the oral notification. If the Region II Administrator cannot be reached, the licensee or other entity shall notify the NRC Operations Center.	

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 75.8(c)	IAEA International Atomic Energy Agency Credential	HBRSEP shall immediately communicate by telephone, within one hour with respect to the credentials of any person who claims to be an IAEA representative and shall accept written or electronic confirmation of the credentials from the NRC.	– If the IAEA representative's credentials have not been confirmed by the NRC, the licensee shall not admit the person until the NRC has confirmed the person's credentials. The licensee, shall notify the Commission promptly, by telephone, whenever an IAEA representative arrives at a facility or location without advance notification.

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
SHUTDOWN REQUIRED BY TS 10 CFR 50.72(b)(2)(i)	Shutdown TS Shutdown Power Reduction	The initiation of any shutdown required by the TS.	– Reactor is in MODEs 1 or 2 and the Control Room takes action to reduce power (i.e., negative reactivity insertion) in order to comply with a Required Action to be in MODE 3 within a Completion Time. Reduction in power for some other purpose than compliance with the shutdown requirement is not reportable. MODE changes required by TS when reactor is in MODEs 3, 4, or other non-power conditions, are not reportable. – If allowed outage time plus required shutdown time to MODE 3 is less than the expected restoration time of the LCO and power is reduced in anticipation of the required shutdown, the shutdown is reportable.
ECCS DISCHARGE INTO RCS 10 CFR 50.72(b)(2)(iv)(A)	ECCS Actuation Safety Injection	Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	– Manual or automatic Safety Injection System actuation in response to a valid signal that resulted in or should have resulted in discharge into the reactor coolant system.
RPS INITIATION (MANUAL/AUTOMATIC) DURING OPERATION 10 CFR 50.72(b)(2)(iv)(B)	RPS Actuation Reactor Protection System RPS Reactor Trip	Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	- Manual or automatic reactor trip from critical through RTP of 100%. Trips which occur as part of planned evolutions in accordance with procedures are not reportable.

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)(xi) 10 CFR 72.75(b)(2)	News Release Press Radio Television Fatality Environment Public Health and Safety Release ISFSI	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 (EPNOT-01). – 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 sources or spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OR THEFT OF LICENSED MATERIAL (>1000X 10 CFR 20 LIMITS) 10 CFR 20.2201	Loss Theft Missing Licensed Radioactive Material Recovery	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. Follow-up written report required within subsequent 30 days. Note – If the lost, stolen, or missing source exceeds a "Quantity of Concern" as specified in HPP-018, then the NRC desires to be notified within 4 hours of any subsequent recovery of the source.	– 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.
ISFSI - DEPARTURE FROM LICENSE CONDITION 10 CFR 72.75(b)(1)	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).

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT 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 eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRINCIPAL SAFETY BARRIERS SERIOUSLY DEGRADED 10 CFR 50.72(b)(3)(ii)(A)	Degraded Safety Barriers Fission Product Barrier	Any event or condition 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 – Low temperature overpressure transients where the pressure temperature limits are violated – 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)(3)(ii)(B)	Safety Function Unanalyzed Condition	Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.	– OTΔT setpoints are declared inoperable due to summator module lag constants. The channel response time exceeded the value assumed in the accident analysis (analytical limits). – 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. – Any power level excursion above 2346 MWt should be evaluated to determine if the condition posed an unanalyzed condition that significantly degrades nuclear plant safety. Operation slightly in excess of 2346 MWt for short periods are not expected to trigger the 10 CFR 50.72(b)(3)(ii)(B) criterion.

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT 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 eight hours 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)(3)(xiii)	Selective Signaling System Sirens ETS ERFIS ERDS	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 15 or more of 59 Public Warning Sirens 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 – ETS communications function unavailable. This does not apply to minor interruptions in site or corporate telecommunications systems. It is intended to apply to serious conditions during which the telecommunication system can no longer fulfill the requirements of the Emergency Plan or provide ETS functionality. **(1)** – Loss of commercial telephone system to the extent that required communications could not be made to official offsite locations (e.g., EOCs, Warning Points) – Inoperability of ERFIS and ERDS is not capable of being restored within one hour. **(2)**

(1) A loss of a single or multiple voice communications (ENS, HPN, or commercial phone) does **NOT** constitute a loss of ETS communications provided alternate telephone communication capability exists. The NRC Operations Center shall be notified as quickly as practical to use alternate means of phone communications. This satisfies the guidance provided in previous Information Notices 85-44 "Emergency Communication System Monthly Test," dated May 30, 1985 and 86-97 "Emergency Communications System," dated November 28, 1986, to test the backup means of communication when the primary system is unavailable as well as the reporting requirements of § 50.72(b)(2)(xii). Loss of either ENS or HPN does not require additional event reporting under 10 CFR 50.72 or 10 CFR 50.73.

(2) **IF** ERDS is inoperable **AND** ERFIS is operable, **THEN** no reportable condition exists

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT 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 eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
RPS/SAFETY SYSTEM INITIATION (MANUAL/AUTOMATIC) 10 CFR 50.72(b)(3)(iv)(A)(B)	Manual Automatic Actuation Engineered Safety Feature ESF Valid Clearance RPS Actuation Reactor Protection System RPS Reactor Trip	Any event or condition that results in valid actuation of any of the systems listed below except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. The systems to which the requirements of this paragraph apply are: (1) Reactor protection system (RPS) including: reactor scram and reactor trip. (2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). (3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems. (4) PWR auxiliary or emergency feedwater system. (5) Containment heat removal and depressurization systems, including containment spray and fan cooler systems. (6) Emergency AC electrical power systems, including: emergency diesel generators (EDGs)	– Auxiliary Feedwater initiation/actuation – Reactor Trip (Manual/Automatic) while subcritical – Reactor Trip while critical is reportable per Attachment 11.2 – EDG start due to an undervoltage trip signal on emergency bus E1 or E2 – A single train of Containment Isolation actuates – A valid signal for Containment Ventilation Isolation occurs Valid actuations are those actuations that result from "valid signals" or from intentional manual initiation, unless it is part of a preplanned test. Valid signals are those signals that are initiated in response to actual plant conditions or parameters satisfying the requirements for the initiation of the safety function of the system. They do not include actuations which are the result of other signals. (NUREG 1022) Invalid actuations are, by definition, those that do not meet the criteria for being valid. Thus invalid actuations include actuations that are not the result of valid signals and are not intentional manual actuations. Except for actuations of the Reactor Protection System (RPS) when the reactor is critical or in MODE 1, invalid actuations are not reportable by telephone under 10 CFR 50.72. In addition, invalid actuations are not reportable under 10 CFR 50.73 in any of the following: - The invalid actuation occurred when the system is already properly removed from service. This means all requirements of plant procedures for removing equipment from service have been met. It includes required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers. - The invalid actuation occurs after the safety function has already been completed. An example would be RPS actuation after the control rods have already been inserted into the core.

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT 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 eight 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)(3)(v)	Loss of Safety Function Residual Heat Mitigation Shutdown Generic Setpoint Drift Engineering Evaluation Operability Determination Common Mode Failure	Any event or condition that at the time of discovery 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. Events covered in this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported in accordance with this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.	– Loss (inoperability) of both Trains, e.g., ECCS, Low Temperature Overpressure Protection System, or Lake Robinson water level below LCO 3.7.8 limit - 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 Control Rod failures (if failure prevented the fulfillment of a safety function) – Operator action to inhibit the RPS (actions would prevent fulfillment of a safety function)

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT 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 eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - DEFECT IMPORTANT TO SAFETY 10 CFR 72.75(c)(1)	ISFSI Defect Safety	A defect in any spent fuel storage structure, system, or component that 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 72.75(c)(2)	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.
TRANSPORT OF CONTAMINATED INJURED PATIENT 10 CFR 50.72(b)(3)(xii) 10 CFR 72.75(c)(3)	Contaminate Injured Person Medical Transport Rescue Hospital ISFSI	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 (NUREG 1022) person to an off-site medical facility for treatment.

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via ETS not later than 24 hours after the discovery of any of the following events or conditions involving spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
FFD SIGNIFICANT EVENT REPORT 10 CFR 26.719	Alcohol Influence Substance NRC employee FFD Fitness for Duty	The use, sale, distribution, possession, or presence of illegal drugs, or the consumption or presence of alcohol within the protected area and, Any acts by any person licensed under 10 CFR part 55 to operate a power reactor or by any supervisory personnel assigned to perform duties within the scope of this part: - involving the sale, use, or possession of a controlled substance, - resulting in confirmed positive tests on such persons, - involving use of alcohol within the protected area, or - resulting in a determination of unfitness for scheduled work due to the consumption of alcohol.	

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC - FFD

See SEC-NGGC-2140 for additional guidance. This will likely be reported by Corporate personnel.

FALSE POSITIVE ERROR ON FFD SPECIMEN 10 CFR 26.719(b)(4)(c)(2)	FFD Fitness for Duty False Positive Specimen Laboratory	Should a false positive error occur on a blind performance test specimen, HBRSEP shall promptly notify the NRC.	See SEC-NGGC-2140 for additional guidance. This will likely be reported by Corporate personnel.
FALSE NEGATIVE ERROR ON FFD SPECIMEN 10 CFR 26.719(b)(4)(c)(3)	FFD Fitness for Duty False Negative Specimen Laboratory	Should a false negative error occur on a quality assurance check of validity screening tests, HBRSEP shall promptly notify the NRC.	

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within twenty-four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - SAFETY EQUIPMENT DISABLED OR FAILURE TO FUNCTION 10 CFR 72.75(d)(1)	ISFSI Disable Failure Safety Function Mitigation Accident Loss of Function	Any event in which important to safety equipment is disabled or fails to function as designed when: i) The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident; and, ii) No redundant equipment was available and operable to perform the required safety function.	

ATTACHMENT 11.4

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

TWENTY-FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within twenty-four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
EXTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, OR SNM (> ANNUAL LIMITS) 10 CFR 20.2202(b)	External Exposure Release Source Byproduct SNM	The discovery of any event involving loss of control of licensed material possessed by HBRSEP that may have caused or threatens to cause any of the following conditions: 1. An individual to receive in a period of 24 hours: (i) A total effective dose equivalent exceeding 5 rems; or (ii) An eye dose equivalent exceeding 15 rems; or (iii) A shallow dose equivalent to the skin or extremities exceeding 50 rems; 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 in excess of one occupational annual limit on intake.	

100. G2.4.43 SRO 001

Which ONE (1) of the following correctly completes the statements below regarding the process for which the NRC Headquarters Operations Officer (HOO) will contact the control room with a notification of a Nuclear Power Plant Attack Message?

The HOO will contact the Unit 2 Control Room and state that he has a Nuclear Power Plant Attack Message and will __(1)__. If the incorrect code is given, IAW OMM-001-4, Communications, the SM or his designee is required to __(2)__.

A. (1) provide the four digit alphanumeric authentication code

(2) receive the message and contact security to validate the threat

B. (1) ask if an authentication code is desired

(2) receive the message and contact security to validate the threat

C✓ (1) provide the four digit alphanumeric authentication code

(2) hang up and immediately call back the HOO

D. (1) ask if an authentication code is desired

(2) hang up and immediately call back the HOO

The correct answer is C.

A. Incorrect - The first part is correct. The second part are the actions if a threat was communicated to the control room via another source. These actions are IAW with Attachment 1 of AOP-034, Security Events. The question specifically states IAW OMM-001-4 with the call coming from the NRC.

B. Incorrect. The first distractor is the actions that the Emergency Communicator would take when contacting the State / Counties when performing an Emergency Notification of an Event declaration. The second part are the actions if a threat was communicated to the control room via another source. These actions are IAW with Attachment 1 of AOP-034, Security Events. The question specifically states IAW OMM-001-4 with the call coming from the NRC.

C. Correct.

D. Incorrect - The first distractor is the actions that the Emergency Communicator would take when contacting the State / Counties when performing an Emergency Notification of an Event declaration. The second part of the answer is correct.

Question 100
Tier 3
K/A Importance Rating - RO 3.2 SRO 3.8

Knowledge of emergency communications systems and techniques.

Reference(s) - Sim/Plant design, OMM-001-4
Proposed References to be provided to applicants during examination - None
Learning Objective - OMM-001-4 Self Study
Question Source - NEW
Question Cognitive Level - F
10 CFR Part 55 Content - 41.10 / 45.13
Comments -

SRO: Recieving emergency communications from the NRC is an SRO only function at RNP.

Progress Energy

I
INFORMATION USE

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PLANT OPERATING MANUAL

VOLUME 3

PART 1

OMM-001-4

COMMUNICATIONS

REVISION 20

9.0 **INSTRUCTIONS**

9.1 **Command and Control**

1. During emergency conditions or anytime that the EALs are being reviewed, all requests for information from the COs should be made through the CRS to assure the communications flow is not confusing for the COs. However, the SM <u>may</u> directly request an available operator to place the VLC Switch to emergency and sound alarms.

2. Emergency Plan announcements should be made from the Shift Manager/Emergency Communicator area if the CRS is involved in directing emergency actions to avoid detracting from efforts to place the plant in a safe condition.

9.2 **NRC Authentication Codes**

1. The NRC will provide a single, four digit alphanumeric code to the main control room during the daily plant status communication check at 0400 hours. The NRC will state the current authentication code and then give the new authentication code. These codes will go into effect each day at 0800 hours and are used to validate the caller identification during imminent threats and physical attacks. In the event of an imminent threat notification between 0400 hours and 0800 hours, the authentication code from the previous day shall be used. The authentication code is not classified as safeguards information but should be recorded and maintained in a discrete location readily accessible to the Shift Manager or designee. The authentication code shall not be entered into Autolog.

2. The call process of reporting an imminent threat is illustrated below:

   a. NRC Headquarters Operations Officer (HOO) calls the Unit 2 Control Room and states that it has an Emergency Aircraft Threat Warning Message or a Nuclear Power Plant Attack Message and gives the authentication code.

   b. The Shift Manager or designee will verify the code and reply that they have confirmed the authentication code and are ready to receive the emergency message.

   c. If an incorrect code is given, the SM or designee shall hang up and immediately call back the HOO. A code word is not required for the call back.

EXAMPLE:

<u>NRC HOO</u>: "This is the NRC Operations Officer, I have an Emergency Aircraft Imminent Threat Warning Message, the authentication code is Alpha One Bravo Yankee"

<u>SM</u>: Checks current authentication code, and if correct, responds: "Authentication confirmed, standing by for warning message, go ahead NRC."

ATTACHMENT 1

CREDIBILITY EVALUATION

(Page 1 of 2)

NOTE

IF at any time during performance of this attachment verification of the threat is received from the NRC with the proper Authentication Code OR Plant Security, THEN the threat is considered credible.

1. IF the caller is from a Federal or State Agency other than the NRC OR the NRC Authentication Code was NOT valid, THEN perform the following:

   a. Obtain the following information from the caller:

      - Name
      - Position/Title
      - Estimated time event will occur

   b. Contact the NRC via NRC ENS phone AND request assistance for verification of threat credibility.

2. IF the caller is NOT from a Federal OR State Agency, THEN attempt to ascertain the following information via questioning the caller:

   - Is the caller rational OR sober?
   - Ask the caller when the event will occur.
   - Does the caller know the specifics concerning the plant?
   - Ask the caller why they are making the call.
   - As a final question, ask the caller his/her name.

3. IF the call is received directly in the Control Room, THEN notify Security that a threat call has been received.

4. If time permits, THEN contact the Plant General Manager OR On-Call Manager for consultation.

Distractor

ATTACHMENT 1

CREDIBILITY EVALUATION

(Page 2 of 2)

5. WHEN a determination for credibility is made, THEN return to the Main Body, Step in effect.

- END -

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM
INSTRUCTIONS FOR COMPLETION

Line 2:

> **CAUTION**
>
> Chronological order of the times recorded on the form is critical. The time listed on Line 2 (NOTIFICATION TIME) should be the **last** time entered on the form. The times on the Electronic Emergency Notification Form should be completed as follows: *first*- Line 10 (DECLARATION TIME), *second*- Line 17 (APPROVAL TIME), and *last*- Line 2 (NOTIFICATION TIME). For example Line 10 at 12:00 and Line 17 at 12:10 and Line 2 at 12:14. The first voice contact (NOTIFICATION TIME) time should not be documented until an approved form is available.

NOTIFICATION TIME/DATE: The Notification Time/Date is completed **when first voice contact is made** during offsite agency notification. First voice contact **will be considered complete after site identification, type of message, and emergency classification is provided to the offsite agency**.

RECORD the time of first voice contact with any offsite agency, as verified on the phone by roll call.

Distractor

AUTHENTICATION #: **ENTER** the **AUTHENTICATION** number in the space provided on the Electronic Emergency Notification Form. This information will be entered after the form is initially developed and transmitted to offsite agencies. Authentication is not required, but the State/County representatives should be asked, "Would anyone like to authenticate this message?" **IF** yes, **THEN** they will pick a number **AND** you respond with the corresponding word (see the authentication code list or select the authentication code button on the electronic emergency notification log).

"GOVERNMENT AGENCIES NOTIFIED"

After review of the ENF, **RECORD** the name of the individual from each agency and the date/time the information was provided.

Upon completion, transmit the information electronically using the "Send Form" button. This will also automatically update the ENF with the earliest notification time.