Form 3.2-1 Administrative Topics Outline

Facility: __Salem_____		Date of Examination: _Feb 13, 2023__
Examination Level: RO ☒ SRO ☐		Operating Test Number: _21-01_____
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	Perform Control Room Logs Modes 1-4 IAW S2.OP-DL.ZZ-0003 [2.1.18, RO–3.6; Ability to make accurate, clear, and concise logs, records, status boards, and reports.]	S, D
Conduct of Operations	Calculate the required boric acid flowrate and time to raise level in the RWST to the Tech Spec minimum [2.1.23, RO-4.3; Ability to perform general and/or normal plant procedures during any plant conditions	R, N
Equipment Control	Perform Seven Day AFD surveillance test IAW S2.OP-ST.NIS-0001 [2.2.12, RO-3.7; Knowledge of surveillance procedures]	R, D
Radiation Control	NOT USED	
Emergency Plan	Activate ERDS during an Emergency Event IAW EP-SA-111-F8 [2.4.43 RO-3.2; Knowledge of emergency communication systems and techniques]	S,D

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	**SRO and RO Retakes**
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)

(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)

(N)ew or Significantly (M)odified from bank (no fewer than one)

SUMMARY:

(RO-A1a) Perform Control Room Logs IAW S2.OP-DL.ZZ-0003:

While reviewing control logs, the CRS discovers that the Reactor Coolant Temperature section of the Control Room Logs was not completed. The applicant will be directed to complete pages 35 and 36. The task is considered satisfactory when the applicant determines that the OPΔT setpoint for 23 Loop is outside the REM criteria (within +/- 1.5°F of REM setpoint) and circles and/or records an UNSAT reading. (Note: This task will be administered in the simulator)

(RO-A1b) Determine Time to Makeup to Unit 2 RWST:

The candidate is directed to determine the boric acid flowrate and the time required to raise the level in the RWST back to the Tech Spec minimum. In order to perform this task, the candidate is directed to perform a manual make-up to the RWST using section 5.8 of S2.OP-SO.CVC-0006, Boron Concentration Control. The candidate will need to utilize the reactor engineering figures procedure S2.RE-RA.ZZ-0012, the normal operating procedure S2.OP-SO.CVC-0006 and the Tank capacity figures in S2.OP-TM.ZZ-0002 to calculate and determine the correct answers to the given questions.

(RO-A2) Perform Seven Day AFD surveillance test IAW S2.OP-ST.NIS-0001:

The candidate is required to manually calculate Axial Flux Difference (AFD) IAW S2.OP-ST.NIS-0001 Power Distribution – Axial Flux Difference. The candidate will review the stem conditions given and utilize that data to perform a calculation of the target AFD and compare that to the actual AFD readings given in the stem. The candidate will determine that one channel exceeds the target value and value is within the acceptable operation region of the Core Operating Limits Report (COLR) boundaries.

(RO-A4) Activate ERDS during an Emergency Event:

The applicant will be directed to activate the ERDS system during an emergency IAW EP-SA-111-F8. The task is considered satisfactory when the applicant successfully activates the ERDS system. (Note: This task will be administered in the simulator)

Form 3.2-1 Administrative Topics Outline

Facility: __Salem_____		Date of Examination: _Feb 13, 2023__
Examination Level: RO ☐ SRO ☒		Operating Test Number: _21-01_____
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	Determine which ECCS pumps can be stopped to maintain minimum required injection flow IAW 2-EOP-LOCA-5, step 19.2, Figure A [2.1.7, SRO-4.7: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.]	R, M
Conduct of Operations	Apply Overtime Limit Restrictions for Emergent Shift Relief [2.1.5, SRO-3.9: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.]	R, D
Equipment Control	Review and approve a completed Containment Ventilation Valve surveillance. [2.2.12, SRO-4.1: Knowledge of surveillance procedures]	R, P
Radiation Control	Evaluate Radiation Hazards and Select a Volunteer for Emergency Radiation Exposure [2.3.14, SRO-3.8: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan of regulatory limits (SRO Only)]	R, D

Emergency Plan	Perform Accuracy Peer Check on an Initial Contact Message Form (ICMF) for a SGTR event [2.4.41, SRO-4.6: Knowledge of the emergency action level thresholds and classifications]	R, N

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	**SRO and RO Retakes**
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)
(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)
(N)ew or Significantly (M)odified from bank (no fewer than one)

SUMMARY:

(SRO-A1a) Determine which ECCS pumps can be stopped:

The candidate will be directed to determine the minimum required injection flow. The candidate will need to choose the pump(s) that can be stopped in accordance with 2-EOP-LOCA-5, Loss of Emergency Recirculation, to maintain minimum required injection flow IAW step 19.2, Figure A.

(SRO-A1b) Apply Overtime Limit Restrictions for Emergent Shift Relief:

The candidate will need to determine the availability of departmental personnel to fulfill an overtime position. The candidate will determine which operator may or may not fill the overtime position that emergently arose and document reasoning IAW LS-AA-119-1003, Calculating Work Hours.

(SRO-A2) Review and approve a completed Containment Ventilation Valve surveillance:

The candidate is required to determine if a completed Containment Ventilation Valve surveillance S2.OP-ST.CBV-0001, Inservice Testing Containment Ventilation Valves Modes 1-6, was completed correctly and can be approved. In this case, the SRO will determine that the surveillance is UNSAT and identify that 2VC13 stroke time is in the Required Action Range and must be declared inoperable.

(SRO-A3) Evaluate Radiation Hazards and Select a Volunteer for Emergency Radiation Exposure:

The candidate will be placed in an emergency situation occurring at the unit with high radiation dose rates in the plant. Based on the radiation levels in the plant, the candidate must select preferred volunteer(s) that will be exposed to a lifesaving radiation dose using the administrative guidelines per NC.EP-EP.ZZ-0304 Operational Support Center (OSC) Radiation Protection Response.

(SRO-A4) Perform Accuracy Peer Check on an Initial Contact Message Form (ICMF) for a SGTR event (Time Critical):

The candidate perform an accuracy peer check on a provided Initial Contact Message Form (ICMF) EP-SA-325-F2, ALERT. They will document any errors on an attached worksheet, should any exist.

Form 3.2-2 Control Room/In-Plant Systems Outline

Facility: _____Salem_____ Date of Examination: ___2/13/2023____
Operating Test Number: _21-01_____

Exam Level: ■ RO ☐ SRO-I ☐ SRO-U

System/JPM Title	Type Code	Safety Function
Control Room Systems		
a. Control Rod exercise surveillance (001, A4.16, 3.8) **Alternate Path** – Control rods continue to insert after placing in Auto & Manual requiring Rx Trip.	A,D,P,S	1
b. Transfer to Hot Leg Recirculation IAW EOP-LOCA-4 (006, A4.05, 4.1) [**MODIFIED** alternate path to start 21 SI pump]	A,EN,L,M,S	2
c. Depressurize RCS using Auxiliary Spray IAW 2-EOP-LOCA-2 (010, A4.05 3.2) **Alternate Path** – Only one PZR PORV is available and does not open or open fully to lower RCS pressure requiring using the alternate path of Aux Spray.	A,L,M,S	3
d. Perform Bleed and Feed IAW 2-EOP-FRHS-1 (002, A2.04, 4.5) **Alternate Path** – Only one PORV opens requiring candidate to open all Rx head vents.	A,D,L,S	4-Pri
e. Depressurizing a SG during EOP-FRHS-1 using Steam Dumps (041, A4.02 3.6).	L,N,S	4-Sec
f. Perform 22 CFCU surveillance test IAW S2.OP-ST.CBV-0003 (022, A1.04, 3.3)	D,S	5
g. Respond to a failed high RCS T-avg channel and defeat the failed channel IAW S2.OP-AB.ROD-0003. (012, A4.05, 3.8)	D,S	7
h. Perform a containment pressure relief IAW S2.OP-SO.CBV-0002. (029, A1.02, 3.2)**Alternate Path** – High radiation alarm fails to isolate containment purge requiring manual actions.	A,EN,N,S	8
In-Plant Systems		

i. Start Station Blackout Air Compressor IAW SC.OP-SO.CA-0001 per Blackout Coping Actions of S2.OP-AB.LOOP-0001. (078, A4.02, 3.2)	D,E,P	8
j. Parallel Rod Control M-G Set IAW S1.OP-SO.RCS-0001 (012, A4.07, 3.8)	D (Unit 1 only)	7
k. Perform a radioactive waste gas release IAW S2.OP-SO.WG-0008 (071, A1.03, 2.6)	D,R	9

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5. The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

- At least one of the tasks shall be related to a shutdown or low-power condition.
- Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures.
- At least one alternate path JPM must be new or modified from the bank.
- At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.
- At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

Code	License Level Criteria		
	RO	SRO-I	SRO-U
(A)lternate path	4–6 (5)	4–6 (4)	2–3 (2)
(C)ontrol room			
(D)irect from bank	≤ 9 (7)	≤ 8 (7)	≤ 4 (3)
(E)mergency or abnormal in-plant	≥ 1 (1)	≥ 1 (1)	≥ 1 (1)
(EN)gineered safety feature (for control room system)	≥ 1 (2)	≥ 1 (1)	≥ 1 (1)
(L)ow power/shutdown	≥ 1 (4)	≥ 1 (4)	≥ 1 (2)
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2 (4)	≥ 2 (3)	≥ 1 (2)
(P)revious two exams (randomly selected)	≤ 3 (2)	≤ 3 (2)	≤ 2 (1)
(R)adiologically controlled area	≥ 1 (1)	≥ 1 (1)	≥ 1 (1)
(S)imulator			

Summary:

Sim-a - Control Rod exercise surveillance:

The candidate will be directed to continue performing control rod surveillance testing. The candidate will complete rod exercise test for Control Bank D by inserting 15 steps. The candidate will verify proper rod insertion and then withdraw rods to original position. When the candidate places control rods in Auto, rods will continuously insert. The candidate will attempt to place rods in manual to stop rod motion, but rods will continue to insert. The candidate will take action to manually trip the reactor.

Sim-b - Perform transfer to Hot leg Recirculation:

The Candidate will be required to perform the Actions of EOP-LOCA-1, Loss of Reactor or Secondary Coolant, to begin the pre-alignment steps for aligning to hot leg recirculation and will transition to EOP-LOCA-4, Transfer to Hot Leg Recirculation, and perform the actions to align for Hot Leg Recirculation with One RHR pump and one SI pump unavailable.

Sim-c – Depressurize RCS using Auxiliary Spray:

The candidate will be directed to perform actions in EOP-LOCA-2, Post-LOCA Cooldown and Depressurization, to depressurize the RCS to refill the Pressurizer at step 11. The candidate will determine that normal spray is not available based on available RCPs. Attempt to open one PZR PORV will be unsuccessful due to a malfunction of the only available PZR PORV (other PZR PORV was isolated). The candidate will use the alternate path of depressurizing the RCS using Auxiliary Spray. The task is considered satisfactory when Aux Spray has been placed in service and RCS pressure is lowering or rising PZR level.

Sim-d – Perform feed and bleed of Reactor Coolant System:

The candidate will be directed to perform actions in EOP-FRHS-1, Response to Loss of Secondary Heat Sink, to perform feed and bleed actions. When performing steps to open the Pressurizer Power Operated Relief Valves one valve will fail to open. The candidate will then perform alternate steps to open the reactor Head vents as an additional bleed path for the RCS.

Sim-e – Depressurize a Steam Generator to feed with Condensate:

The candidate will be directed to perform actions in EOP-FRHS-1, Response to Loss of Secondary Heat Sink, to perform steam generator depressurization. The candidate will perform steps to depressurize the steam generator by isolating three of four steam generators and operating steam dumps to depressurize the individual steam generator, and aligning the feed system to fill the steam generator using the condensate pumps. Following the depressurization the candidate will place steam dumps in auto to stabilize pressure.

Sim-f – Perform 22 Containment Fan Coil Unit Surveillance testing:

The candidate will be directed to perform actions in S2.OP-ST.CBV-0003, Containment systems – Cooling systems, to perform the 22 Containment Fan Coil Unit (CFCU) surveillance. The candidate will perform the surveillance by securing the 22 CFCU and restarting it in Slow speed,

the SW outlet valve the 22SW223 will have a malfunction entered resulting in lower than normal flow from the 22 CFCU. When performing steps to record the required differential pressure data the candidate will complete the surveillance determining that the data reported from the field is unsatisfactory, the surveillance is unsatisfactory and the 22 CFCU is inoperable.

Sim-g – Respond to a Failed High RCS Tavg Channel and Defeat the Channel:

The candidate will response to alarms and continuous rod motion due to a failed RCS T-avg channel. The candidate will enter S2.OP-AB.ROD-0003, Continuous Rod Motion, and take the action to place rods in manual, place the charging Master Flow Controller in manual to stabilize PZR level, and then defeat the failed T-avg channel. The task is considered satisfactory when the rods are in manual, PZR level is stable, and the T-avg channel is defeated.

Sim-h – Perform a containment pressure relief:

The candidate will be directed to perform a containment pressure relief IAW S2.OP-SO.CBV-0002, Containment Pressure-Vacuum Relief System Operation, to lower containment pressure. The candidate will perform steps to place the pressure relief in service. Upon completion of the valve alignment and opening of the pressure relief damper the 2R41, Plant Vent Radiation Monitor indications will begin to rise. The rising levels will eventually result in a Containment Ventilation Isolation signal being generated. The dampers that were realigned to perform the pressure relief will fail to automatically close from the signal and the candidate will be required to manually terminate the ongoing release.

In Plant-i – Start Station Blackout Air Compressor:

The candidate will be directed to perform blackout coping actions in S2.OP-AB.LOOP-0001, Loss of Off-Site Power, to start the Station Blackout Air Compressor per SC.OP-SO.CA-0001, SBO Diesel Control Air Compressor, due to station blackout conditions. This JPM meets the requirement for performing an in-plant task during an Emergency or Abnormal plant condition.

In Plant-j – Parallel rod control motor generator set:

The candidate will be directed to perform actions in S1.OP-SO.RCS-0001, Rod Control System Operation, to parallel the second motor generator set for the rod control system. The candidate will simulate performing switch manipulations in the field to perform the paralleling of the motor generator set.

In Plant-k – Perform a radioactive waste gas release:

The candidate will be directed to perform a radioactive waste gas release IAW S2.OP-SO.WG-0008, Discharge of 21 Gas Decay Tank to Plant Vent. The candidate will simulate performing steps to discharge the 21 waste gas decay tank to the plant vent. This JPM is located in the radiologically controlled area of the auxiliary building.

Form 3.2-2 Control Room/In-Plant Systems Outline

Facility: _____Salem_____ Date of Examination: ___2/13/2023____
Operating Test Number: _21-01_____

Exam Level: ☐ RO ■ SRO-I ☐ SRO-U

System/JPM Title	Type Code	Safety Function
Control Room Systems		
a. Control Rod exercise surveillance (001, A4.16, 3.8) **Alternate Path** – Control rods continue to insert after placing in Auto & Manual requiring Rx Trip.	A,D,P,S	1
b. Transfer to Hot Leg Recirculation IAW EOP-LOCA-4 (006, A4.05, 4.1) [**MODIFIED** alternate path to start 21 SI pump]	A,EN,L,M,S	2
c. Depressurize RCS using Auxiliary Spray IAW 2-EOP-LOCA-2 (010, A4.05 3.2) **Alternate Path** – Only one PZR PORV is available and does not open or open fully to lower RCS pressure at step 10 requiring using the alternate path of Aux Spray.	A,L,M,S	3
d. Perform Bleed and Feed IAW 2-EOP-FRHS-1 (002, A2.04, 4.5) **Alternate Path** – Only one PORV opens requiring candidate to open all Rx head vents.	A,D,L,S	4-Pri
e. Depressurizing a SG during EOP-FRHS-1 using Steam Dumps (041, A4.02 3.6).	L,N,S	4-Sec
f. Perform 22 CFCU surveillance test IAW S2.OP-ST.CBV-0003 (022, A1.04, 3.3)	D,S	5
g. Respond to a failed high RCS T-avg channel and defeat the failed channel IAW S2.OP-AB.ROD-0003. (012, A4.05, 3.8)	D,S	7
h. Not Used		
In-Plant Systems		

i. Start Station Blackout Air Compressor IAW SC.OP-SO.CA-0001 per Blackout Coping Actions of S2.OP-AB.LOOP-0001. (078, A4.02, 3.2)	D,E,P	8
j. Parallel Rod Control M-G Set IAW S1.OP-SO.RCS-0001 (012, A4.07, 3.8)	D (Unit 1 only)	7
k. Perform a radioactive waste gas release IAW S2.OP-SO.WG-0008 (071, A1.03, 2.6)	D,R	9

Form 3.2-2 Control Room/In-Plant Systems Outline

Facility: _____Salem_____ Date of Examination: ___2/13/2023____
Operating Test Number: _21-01_____

Exam Level: ☐ RO ☐ SRO-I ■ SRO-U

System/JPM Title	Type Code	Safety Function
Control Room Systems		
b. Transfer to Hot Leg Recirculation IAW EOP-LOCA-4 (006, A4.05, 4.1) [**MODIFIED** alternate path to start 21 SI pump]	A,EN,L,M,S	2
c. Depressurize RCS using Auxiliary Spray IAW 2-EOP-LOCA-2 (010, A4.05 3.2) **Alternate Path** – Only one PZR PORV is available and does not open or open fully to lower RCS pressure at step 10 requiring using the alternate path of Aux Spray.	A,L,M,S	3
g. Respond to a failed high RCS T-avg channel and defeat the failed channel IAW S2.OP-AB.ROD-0003. (012, A4.05, 3.8)	D,S	7
In-Plant Systems		
i. Start Station Blackout Air Compressor IAW SC.OP-SO.CA-0001 per Blackout Coping Actions of S2.OP-AB.LOOP-0001. (078, A4.02, 3.2)	D,E,P	8
k. Perform a radioactive waste gas release IAW S2.OP-SO.WG-0008 (071, A1.03, 2.6)	D,R	9

Form 4.1-PWR Pressurized-Water Reactor Examination Outline

Facility: Salem													**K/A Catalog Rev. 3**		**Rev.** 0		**Date of Exam:** 2/13/2023		
Tier	Group	RO K/A Category Points													**SRO-Only Points**				
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total		**A2**		**G***		**Total**
1. Emergency and Abnormal Plant Evolutions	1	3	3	3				3	3			3	18		**3**		**3**		**6**
	2	1	1	2				2	1			1	8		**2**		**2**		**4**
	Tier Totals	4	4	5				5	4			4	26		**5**		**5**		**10**
2. Plant Systems	1	3	3	3	3	2	1	2	3	3	2	3	28		**3**		**2**		**5**
	2	1	0	1	0	1	2	1	1	0	1	1	9		**0**	**2**	**1**		**3**
	Tier Totals	4	3	4	3	3	3	3	4	3	3	4	37		**5**		**3**		**8**

3. Generic Knowledge and Abilities Categories	CO	EC	RC	EM		**CO**	**EC**	**RC**	**EM**	
	2	2	1	1	6	**2**	**2**	**1**	**2**	**7**

4. Theory	Reactor Theory	Thermodynamics	
	3	3	6

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control;
EM = Emergency Procedures/Plan

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan.

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan.

ES-4.1-PWR

PWR Examination Outline (Salem)

Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)

Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
1	(000007) (EPE 7; BW E02 & E10; CE E02) Reactor Trip, Stabilization, Recovery					X		(000007EA2.08) Ability to determine and/or interpret the following as they apply to (EPE 7) REACTOR TRIP, STABILIZATION, RECOVERY (CFR: 41.10 / 43.5 / 45.13): RCS loop flow rates	3.0	1
2	(000009) (EPE 9) Small Break LOCA	X						(000009EK1.03) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (EPE 9) SMALL-Break LOCA (CFR: 41.5 / 41.7 / 45.7 / 45.8): RCS heat removal	4.0	2
3	(000022) (APE 22) Loss of Reactor Coolant Makeup				X			(000022AA1.05) Ability to operate and/or monitor the following as they apply to (APE 22) LOSS OF REACTOR Coolant Makeup (CFR: 41.5 / 41.7 / 45.5 to 45.8): RCP seal backpressure regulator valves	3.3	3
4	(000026) (APE 26) Loss of Component Cooling Water						X	(000026) (APE 26) Loss of Component Cooling Water (G2.1.43) CONDUCT OF OPERATIONS: Ability to use an online power distribution monitoring system and/or procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, or fuel depletion	4.1	4
5	(000027) (APE 27) Pressurizer Pressure Control System Malfunction					X		(000027AA2.13) Ability to determine and/or interpret the following as they apply to (APE 27) PRESSURIZER PRESSURE Control System Malfunction (CFR: 41.10 / 43.5 / 45.13): Seal return flow	3.3	5
6	(000038) (EPE 38) Steam Generator Tube Rupture						X	(000038) (EPE 38) Steam Generator Tube Rupture (G2.1.3) CONDUCT OF OPERATIONS: Knowledge of shift or short-term relief turnover practices (CFR: 41.10 / 45.13)	3.7	6
7	(000040) (APE 40; BW E05; CE E05; W E12) Steam Line Rupture – Excessive Heat Transfer		X					(WE12EK2.03) Knowledge of the relationship between (W E12) UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS and the following systems or components (CFR: 41.8 / 41.10 / 45.3): AC electrical distribution system	3.2	7
8	(000054) (APE 54; CE E06) Loss of Main Feedwater						X	(000054) (APE 54; CE E06) Loss of Main Feedwater (G2.3.12) RADIATION CONTROL: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters (CFR: 41.12 / 43.4 / 45.9 / 45.10)	3.2	8
9	(000055) (EPE 55) Station Blackout			X				(000055EK3.01) Knowledge of the reasons for the following responses and/or actions as they apply to (EPE 55) Station Blackout (CFR: 41.5 / 41.10 / 45.6 / 45.13): Length of time for which battery capacity is designed	4.1	9
10	(000056) (APE 56) Loss of Offsite Power			X				(000056AK3.02) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 56) Loss of Offsite Power (CFR: 41.5 / 41.10 / 45.6 / 45.13): Actions contained in AOPs	4.1	10
11	(000057) (APE 57) Loss of Vital AC Instrument Bus		X					(000057AK2.08) Knowledge of the relationship between (APE 57) LOSS OF VITAL AC ELECTRICALINSTRUMENT BUS and the following systems or components (CFR: 41.8 / 41.10 / 45.3): NI	4.1	11
12	(000058) (APE 58) Loss of DC Power			X				(000058AK3.01) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 58) LOSS OF DC Power (CFR: 41.5 / 41.10 / 45.6 / 45.13): Operation of the EDGs	4.0	12
13	(000062) (APE 62) Loss of Nuclear Service Water		X					(000062AK2.08) Knowledge of the relationship between (APE 62) LOSS OF SERVICE WATER and the following systems or components (CFR: 41.8 / 41.10 / 45.3): PRMS	3.0	13
14	(000065) (APE 65) Loss of Instrument Air				X			(000065AA1.03) Ability to operate and/or monitor the following as they apply to (APE 65) LOSS OF Instrument Air (CFR: 41.5 / 41.7 / 45.5 to 45.8): Restoration of systems served by instrument air when pressure is regained	3.1	14

15	(000077) (APE 77) Generator Voltage and Electric Grid Disturbances					X		(000077AA2.05) Ability to determine and/or interpret the following as they apply to (APE 77) GENERATOR VOLTAGE AND ELECTRIC Grid Disturbances (CFR: 41.10 / 43.5 / 45.13): Status of grid	3.7	15
16	(W E04) LOCA Outside Containment				X			(WE04EA1.04) Ability to operate and/or monitor the following as they apply to (W E04) LOCA Outside Containment (CFR: 41.5 / 41.7 / 45.5 to 45.8): RCS	3.8	16
17	(W E11) Loss of Emergency Coolant Recirculation	X						(WE11EK1.08) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (W E11) LOSS OF EMERGENCY Coolant Recirculation (CFR: 41.5 / 41.7 / 45.7 / 45.8): RTD bypass manifold temperature inaccuracy on natural circulation	3.2	17
18	(BW E04; W E05) Inadequate Heat Transfer – Loss of Secondary Heat Sink	X						(WE05EK1.09) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (W E05) Loss of Secondary Heat Sink (CFR: 41.5 / 41.7 / 45.7 / 45.8): MSLI on high steam pressure rate during cooldown	3.6	18
19	**(000008) (APE 8) Pressurizer Vapor Space Accident**					**X**		**(000008AA2.15) Ability to determine and/or interpret the following as they apply to (APE 8) PRESSURIZER VAPOR Space Accident (CFR: 41.10 / 43.5 / 45.13): ECCS status**	**3.9**	**76**
20	**(000011) (EPE 11) Large Break LOCA**						**X**	**(000011) (EPE 11) Large Break LOCA (G2.4.14) EMERGENCY PROCEDURES/PLAN: Knowledge of general guidelines for emergency and abnormal operating procedures usage (CFR: 41.10 / 43.1 / 45.13)**	**4.5**	**77**
21	**(000015) (APE 15) Reactor Coolant Pump Malfunctions**					**X**		**(000015AA2.16) Ability to determine and/or interpret the following as they apply to (APE 15) REACTOR COOLANT Pump Malfunctions (CFR: 41.10 / 43.5 / 45.13): RCP vibration**	**3.5**	**78**
22	**(000025) (APE 25) Loss of Residual Heat Removal System**						**X**	**(000025) (APE 25) Loss of Residual Heat Removal System (G2.1.1) CONDUCT OF OPERATIONS: Knowledge of conduct of operations requirements (CFR: 41.10 / 43.10 / 45.13)**	**4.2**	**79**
23	**(000029) (EPE 29) Anticipated Transient Without Scram**					**X**		**(000029EA2.09) Ability to determine and/or interpret the following as they apply to (EPE 29) ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) (CFR: 41.10 / 43.5 / 45.13): Occurrence of a main turbine trip**	**4.1**	**80**
24	**(000054) (APE 54; CE E06) Loss of Main Feedwater**						**X**	**(000054) (APE 54; CE E06) Loss of Main Feedwater (G2.1.25) CONDUCT OF OPERATIONS: Ability to interpret reference materials, such as graphs, curves, and tables (reference potential) (CFR: 41.10 / 43.5 / 45.12)**	**4.2**	**81**
K/A Category Totals:		3	3	3	3	6	6	Group Point Total:		24

ES-4.1-PWR	PWR Examination Outline (Salem)									
	Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)									
Item #	E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
25	(000024) (APE 24) Emergency Boration					X		(000024AA2.09) Ability to determine and/or interpret the following as they apply to (APE 24) EMERGENCY Boration (CFR: 41.10 / 43.5 / 45.13): RWST level	3.0	19
26	(000051) (APE 51) Loss of Condenser Vacuum	X						(000051AK1.02) Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to (APE 51) LOSS OF Condenser Vacuum (CFR: 41.5 / 41.7 / 45.7 / 45.8): Relationship of condenser and MT/G operation	3.7	20
27	(000069) (APE 69; W E14) Loss of Containment Integrity				X			(WE14EA1.08) Ability to operate and/or monitor the following as they apply to (W E14) HIGH CONTAINMENT PRESSURE (CFR: 41.5 / 41.7 / 45.5 to 45.8): CCS	4.0	21
28	(000076) (APE 76) High Reactor Coolant Activity			X				(000076AK3.03) Knowledge of the reasons for the following responses and/or actions as they apply to (APE 76) HIGH REACTOR COOLANT ACTIVITY (CFR: 41.5 / 41.10 / 45.6 / 45.13): Reducing letdown flow rates	2.9	22
29	(W E01 & E02) Rediagnosis & SI Termination						X	(W E01 & E02) Rediagnosis & SI Termination (G2.4.19) EMERGENCY PROCEDURES/PLAN: Knowledge of emergency and abnormal operating procedures layout, symbols, and icons (CFR: 41.10 / 45.13)	3.4	23
30	(W E15) Containment Flooding		X					(WE15EK2.03) Knowledge of the relationship between (W E15) Containment Flooding and the following systems or components (CFR: 41.8 / 41.10 / 45.3): Service water	3.3	24
31	(W E16) High Containment Radiation				X			(WE16EA1.04) Ability to operate and/or monitor the following as they apply to (W E16) HIGH Containment Radiation (CFR: 41.5 / 41.7 / 45.5 to 45.8): Containment ventilation isolation	3.7	25
32	(BW E09; CE A13**; W E09 & E10) Natural Circulation			X				(WE09EK3.06) Knowledge of the reasons for the following responses and/or actions as they apply to (W E09) NATURAL CIRCULATION OPERATIONS (CFR: 41.5 / 41.10 / 45.6 / 45.13): Starting an RCP	3.4	26
33	**(000033) (APE 33) Loss of Intermediate Range Nuclear Instrumentation**					**X**		**(000033AA2.01) Ability to determine and/or interpret the following as they apply to (APE 33) LOSS OF INTERMEDIATE RANGE Nuclear Instrumentation (CFR: 41.10 / 43.5 / 45.13): Equivalency among source range, intermediate range, and power range channel readings**	**3.4**	**82**
34	**(000068) (APE 68; BW A06) Control Room Evacuation**						**X**	**(000068) (APE 68; BW A06) Control Room Evacuation (G2.4.20) EMERGENCY PROCEDURES/PLAN: Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes (CFR: 41.10 / 43.5 / 45.13)**	**4.3**	**83**
35	**(000074) (EPE 74; W E06 & E07) Inadequate Core Cooling**					**X**		**(000074EA2.12) Ability to determine and/or interpret the following as they apply to (EPE 74) Inadequate Core Cooling (CFR: 41.10 / 43.5 / 45.13): RCS cooldown rate**	**3.8**	**84**
36	**(W E13) Steam Generator Overpressure**						**X**	**(W E13) Steam Generator Overpressure (G2.4.28) EMERGENCY PROCEDURES/PLAN: Knowledge of procedures relating to a security event (ensure that the test item includes no safeguards information) (CFR: 41.10 / 43.5 / 45.13)**	**4.1**	**85**
	000001 (APE 1) Continuous Rod Withdrawal / 1									
	000003 (APE 3) Dropped Control Rod / 1									
	000005 (APE 5) Inoperable/Stuck Control Rod / 1									

	000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
	000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
	000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
	000037 (APE 37) Steam Generator Tube Leak / 3									
	000059 (APE 59) Accidental Liquid Radwaste Release / 9									
	000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
	000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
	000067 (APE 67) Plant Fire On Site / 8									
	000078 (APE 78*) RCS Leak / 3									
	(BW A01) Plant Runback / 1									
	(BW A02 & A03) Loss of NNI-X/Y/7									
	(BW A04) Turbine Trip / 4									
	(BW A05) Emergency Diesel Actuation / 6									
	(BW A07) Flooding / 8									
	(BW E03) Inadequate Subcooling Margin / 4									
	(BW E08; W E03) LOCA Cooldown – Depressurization / 4									
	(BW E13 & E14) EOP Rules and Enclosures									
	(CE A11**; W E08) RCS Overcooling – Pressurized Thermal Shock / 4									
	(CE A16) Excess RCS Leakage / 2									
	(CE E09) Functional Recovery									
	(CE E13*) Loss of Forced Circulation / LOOP / Blackout / 4									
K/A Category Totals:		1	1	2	2	3	3	Group Point Total:		12

ES-4.1-PWR	PWR Examination Outline (Salem)														
	Plant Systems—Tier 2/Group 1 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
37	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM											X	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM (G2.1.7) CONDUCT OF OPERATIONS: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation (CFR: 41.5 / 45.12, 45.13)	4.4	27
38	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM									X			(004A3.01) Ability to monitor automatic features of the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM, including (CFR: 41.7 / 45.7): Water and boron inventory	3.7	28
39	(004) (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM			X									(004K3.13) Knowledge of the effect that a loss or malfunction of the (SF1; SF2 CVCS) CHEMICAL AND VOLUME CONTROL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): ECCS	3.8	29
40	(005) (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM										X		(005A4.02) Ability to manually operate and/or monitor the (SF4P RHR) RESIDUAL HEAT REMOVAL SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): RHR heat exchanger temperature/bypass control valves	3.9	30
41	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM					X							(006K5.07) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM (CFR: 41.5 / 45.3): Expected temperature values in various locations of the RCS due to different break locations during all ECCS injection modes	3.1	31
42	(007) (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM	X											(007K1.06) Knowledge of the physical connections and/or cause and effect relationships between the (SF5 PRTS) PRESSURIZER RELIEF/QUENCH TANK SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): LRS	2.6	32
43	(008) (SF8 CCW) COMPONENT COOLING WATER SYSTEM					X							(008K5.10) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF8 CCW) COMPONENT COOLING WATER SYSTEM (CFR: 41.5 / 45.3): Requirements on and for the CCWS for different conditions of the power plant	3.1	33
44	(010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM				X								(010K4.03) Knowledge of (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Overpressure control	3.9	34

45	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM									X			(012A3.06) Ability to monitor automatic features of the (SF7 RPS) REACTOR PROTECTION SYSTEM, including (CFR: 41.7 / 45.7): Trip logic	4.1	35
46	(012) (SF7 RPS) REACTOR PROTECTION SYSTEM	X											(012K1.09) Knowledge of the physical connections and/or cause and effect relationships between the (SF7 RPS) REACTOR PROTECTION SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): RCPS	3.8	36
47	(013) (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM							X					(013A1.14) Ability to predict and/or monitor changes in parameters associated with operation of the (SF2 ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM, including (CFR: 41.5 / 45.5): EDG	3.8	37
48	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM								X				(022A2.07) Ability to (a) predict the impacts of the following on the (SF5 CCS) CONTAINMENT COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): ESFAS actuation	4.0	38
49	(022) (SF5 CCS) CONTAINMENT COOLING SYSTEM		X										(022K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF5 CCS) CONTAINMENT COOLING SYSTEM CCS fans	3.6	39
50	(026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM		X										(026K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF5 CSS) CONTAINMENT SPRAY SYSTEM Containment spray pumps	3.9	40
51	(039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM			X									(039K3.03) Knowledge of the effect that a loss or malfunction of the (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): AFW system	3.3	41
52	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM											X	(059) (SF4S MFW) MAIN FEEDWATER SYSTEM (G2.4.2) EMERGENCY PROCEDURES/PLAN: Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions (CFR: 41.7 / 45.8)	4.5	42
53	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM								X				(061A2.07) Ability to (a) predict the impacts of the following on the (SF4S AFW) AUXILIARY/EMERGENCY FEEDWATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Air-operated valve, solenoid-operated valve, or motor-operated valve failure	4.0	43

54	(061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM											X	**COMPONENT: 191001 K1.04**-The failed-valve positions for different operators (open, closed, and as-is positions' spring-loaded valves; hydraulic, pneumatically controlled valves; electric motor-driven valves).	3.2	44
55	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM								X				(062A2.25) Ability to (a) predict the impacts of the following on the (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Fault on a motor control center	3.3	45
56	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM	X											(063K1.01) Knowledge of the physical connections and/or cause and effect relationships between the (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): Ground detection system	2.6	46
57	(063) (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM				X								(063K4.01) Knowledge of (SF6 ED DC) DC ELECTRICAL DISTRIBUTION SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Manual/automatic transfers of control	3.4	47
58	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM		X										(064K2.01) Knowledge of electrical power supplies to the following (CFR: 41.7): (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM Starting air compressor	3.0	48
59	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM										X		(073A4.02) Ability to manually operate and/or monitor the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): RMS control panel	3.6	49
60	(076) (SF4S SW) SERVICE WATER SYSTEM							X					(076A1.03) Ability to predict and/or monitor changes in parameters associated with operation of the (SF4S SW) SERVICE WATER SYSTEM, including (CFR: 41.5 / 45.5): SWS header pressures	3.3	50
61	(076) (SF4S SW) SERVICE WATER SYSTEM			X									(076K3.16) Knowledge of the effect that a loss or malfunction of the (SF4S SW) SERVICE WATER SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): AFW	3.4	51
62	(078) (SF8 IAS) INSTRUMENT AIR SYSTEM									X			(078A3.05) Ability to monitor automatic features of the (SF8 IAS) INSTRUMENT AIR SYSTEM, including (CFR: 41.7 / 45.7): Isolation of instrument air to containment	3.3	52
63	(103) (SF5 CNT) CONTAINMENT SYSTEM				X								(103K4.05) Knowledge of (SF5 CNT) CONTAINMENT SYSTEM design features and/or interlocks that provide for the following (CFR: 41.7): Containment construction	2.8	53

64	(103) (SF5 CNT) CONTAINMENT SYSTEM						X						(103K6.15) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF5 CNT) CONTAINMENT SYSTEM (CFR: 41.7 / 45.7): MRSS	3.1	54
65	(003) (SF4P RCP) REACTOR COOLANT PUMP SYSTEM								X				(003A2.01) Ability to (a) predict the impacts of the following on the (SF4P RCP) REACTOR COOLANT PUMP SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Problems with RCP seals, especially seal leakoff rates	4.0	86
66	(006) (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM								X				(006A2.12) Ability to (a) predict the impacts of the following on the (SF2; SF3 ECCS) EMERGENCY CORE COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Conditions requiring actuation of ECCS	4.5	87
67	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM											X	(062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM (G2.1.6) CONDUCT OF OPERATIONS: Ability to manage the control room crew during plant transients (SRO Only) (CFR: 43.5 / 45.12 / 45.13)	4.8	88
68	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM											X	(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM (G2.1.47) CONDUCT OF OPERATIONS: Ability to direct nonlicensed personnel activities inside the control room (CFR: 41.10 / 43.5 / 45.5 / 45.12 / 45.13)	3.2	89
69	(073) (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM								X				(073A2.01) Ability to (a) predict the impacts of the following on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): PRM component failures	3.1	90
	025 (SF5 ICE) ICE CONDENSER SYSTEM														
	053 (SF1; SF4P ICS*) INTEGRATED CONTROL SYSTEM														
K/A Category Totals:		3	3	3	3	2	1	2	6	3	2	5	Group Point Total:		33

ES-4.1-PWR	PWR Examination Outline (Salem)														
	Plant Systems—Tier 2/Group 2 (RO/**SRO**)														
Item #	System / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
70	(001) (SF1 CRDS) CONTROL ROD DRIVE SYSTEM	X											(001K1.03) Knowledge of the physical connections and/or cause and effect relationships between the (SF1 CRDS) CONTROL ROD DRIVE SYSTEM and the following systems (CFR: 41.2 to 41.9 / 45.7 to 45.8): CRDM	3.7	55
71	(014) (SF1 RPI) ROD POSITION INDICATION SYSTEM								X				(014A2.11) Ability to (a) predict the impacts of the following on the (SF1 RPI) ROD POSITION INDICATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Failed group demand position indication	3.0	56
72	(017) (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM						X						(017K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF7 ITM) IN CORE TEMPERATURE MONITOR SYSTEM (CFR: 41.7 / 45.7): Temperature measuring device (for example thermocouple)	3.2	57
73	(028) (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM			X									(028K3.02) Knowledge of the effect that a loss or malfunction of the (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM will have on the following systems or system parameters (CFR: 41.7 / 45.4): CNT	2.9	58
74	(033) (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM						X						(033K6.11) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the (SF8 SFPCS) SPENT FUEL POOL COOLING SYSTEM (CFR: 41.7 / 45.7): CCWS	3.4	59
75	(056) (SF4S CDS) CONDENSATE SYSTEM											X	(056) (SF4S CDS) CONDENSATE SYSTEM (G2.4.46) EMERGENCY PROCEDURES/PLAN: Ability to verify that the alarms are consistent with the plant conditions (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.2	60
76	(068) (SF9 LRS) LIQUID RADWASTE SYSTEM					X							(068K5.07) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the (SF9 LRS) LIQUID RADWASTE SYSTEM (CFR: 41.5 / 45.3): Loss of secondary mixing water system	2.6	61
77	(072) (SF7 ARM) AREA RADIATION MONITORING SYSTEM										X		(072A4.01) Ability to manually operate and/or monitor the (SF7 ARM) AREA RADIATION MONITORING SYSTEM in the control room (CFR: 41.7 / 45.5 to 45.8): Alarm and interlock setpoint checks and adjustments	3.3	62

78	(086) (SF8 FPS) FIRE PROTECTION SYSTEM							X					(086A1.06) Ability to predict and/or monitor changes in parameters associated with operation of the (SF8 FPS) FIRE PROTECTION SYSTEM, including (CFR: 41.5 / 45.5): Lights and alarms	3.0	63
79	**(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM**											**X**	**(002) (SF2; SF4P RCS) REACTOR COOLANT SYSTEM (G2.2.37) EQUIPMENT CONTROL: Ability to determine operability or availability of safety-related equipment (SRO Only) (CFR: 43.2 / 43.5 / 45.12)**	**4.6**	**91**
80	**(016) (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM**								**X**				**(016A2.01) Ability to (a) predict the impacts of the following on the (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Detector/transmitter failure**	**3.4**	**92**
81	**(029) (SF8 CPS) CONTAINMENT PURGE SYSTEM**								**X**				**(029A2.01) Ability to (a) predict the impacts of the following on the (SF8 CPS) CONTAINMENT PURGE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 41.5 / 45.6): Maintenance or other activity taking place inside containment**	**2.8**	**93**
	011 (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM														
	015 (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM														
	027 (SF5 CIRS) CONTAINMENT IODINE REMOVAL SYSTEM														
	034 (SF8 FHS) FUEL HANDLING EQUIPMENT SYSTEM														
	035 (SF4P SG) STEAM GENERATOR SYSTEM														
	041 (SF4S SDS) STEAM DUMP / TURBINE BYPASS CONTROL SYSTEM														
	045 (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM														
	050 (SF9 CRV*) CONTROL ROOM VENTILATION														

	055 (SF4S CARS) CONDENSER AIR REMOVAL SYSTEM														
	071 (SF9 WGS) WASTE GAS DISPOSAL SYSTEM														
	075 (SF8 CW) CIRCULATING WATER SYSTEM														
	079 (SF8 SAS**) STATION AIR SYSTEM														
K/A Category Totals:		1	0	1	0	1	2	1	3	0	1	2	Group Point Total:		12

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Salem)						
Facility:	Salem				Date of Exam:	2/13/2023	
Generic Knowledge and Abilities Outline (Tier 3) (RO/SRO)							
				RO		SRO-Only	
Category	K/A #	Topic	Item #	IR	Q#	IR	Q#
1. Conduct of Operations	G2.1.19	(G2.1.19) CONDUCT OF OPERATIONS: Ability to use available indications to evaluate system or component status (CFR: 41.10 / 45.12)	82	3.9	64		
	G2.1.31	(G2.1.31) CONDUCT OF OPERATIONS: Ability to locate control room switches, controls, and indications and to determine whether they correctly reflect the desired plant lineup (CFR: 41.10 / 45.12)	83	4.6	65		
	G2.1.5	**(G2.1.5) CONDUCT OF OPERATIONS: Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential) (CFR: 41.10 / 43.5 / 45.12)**	**84**			**3.9**	**94**
	G2.1.46	**(G2.1.46) CONDUCT OF OPERATIONS: Ability to use integrated control systems to operate plant systems or components (CFR: 41.10 / 45.12 / 45.13)**	**85**			**3.3**	**95**
	Subtotal			N/A	2	N/A	**2**
2. Equipment Control	G2.2.3	(G2.2.3) EQUIPMENT CONTROL: (Multi-unit license) Knowledge of the design, procedural, and/or operational differences between units (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)	86	3.8	66		
	G2.2.14	(G2.2.14) EQUIPMENT CONTROL: Knowledge of the process for controlling equipment configuration or status (CFR: 41.10 / 43.3 / 45.13)	87	3.9	67		
	G2.2.19	**(G2.2.19) EQUIPMENT CONTROL: Knowledge of maintenance work order requirements (CFR: 41.10 / 43.5 / 45.13)**	**88**			**3.4**	**96**
	G2.2.44	**(G2.2.44) EQUIPMENT CONTROL: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions (CFR: 41.5 / 43.5 / 45.12)**	**89**			**4.4**	**97**
	Subtotal			N/A	2	N/A	**2**
3. Radiation Control	G2.3.11	(G2.3.11) RADIATION CONTROL: Ability to control radiation releases (CFR: 41.11 / 43.4 / 45.10)	90	3.8	68		
	G2.3.6	**(G2.3.6) RADIATION CONTROL: Ability to approve liquid or gaseous release permits (CFR: 41.13 / 43.4 / 45.10)**	**91**			**3.8**	**98**
	Subtotal			N/A	1	N/A	**1**
4. Emergency Procedures / Plan	G2.4.29	(G2.4.29) EMERGENCY PROCEDURES/PLAN: Knowledge of the emergency plan implementing procedures (CFR: 41.10 / 43.5 / 45.11)	92	3.1	69		
	G2.4.2	**(G2.4.2) EMERGENCY PROCEDURES/PLAN: Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions (CFR: 41.7 / 45.7 / 45.8)**	**93**			**4.6**	**99**
	G2.4.25	**(G2.4.25) EMERGENCY PROCEDURES/PLAN: Knowledge of fire protection procedures (CFR: 41.10 / 43.5 / 45.13)**	**94**			**3.7**	**100**
	Subtotal			N/A	1	N/A	**2**
Tier 3 Point Total				N/A	**6**	N/A	**7**

Form 4.1-COMMON Common Examination Outline

ES-4.1-COMMON	COMMON Examination Outline (Salem)				
Facility:	Salem			Date of Exam:	2/13/2023
Theory (Tier 4) (RO)					
				RO	
Category	K/A #	Topic	Item #	IR	Q#
Reactor Theory	192006	(192006K1.07) FISSION PRODUCT POISONS (CFR: 41.1): Describe the following processes and state their effect on reactor operations: -- xenon following a trip	95	3.4	70
	192007	(192007K1.05) FUEL DEPLETION AND BURNABLE POISONS (CFR: 41.1): Describe the effects of boration/dilution on reactivity during forced-flow and natural circulation conditions	96	3.2	71
	192008	(192008K1.03) REACTOR OPERATIONAL PHYSICS (CFR: 41.1): (STARTUP AND APPROACH TO CRITICALITY) Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality	97	4	72
	Subtotal			N/A	3
Thermodynamics	193003	(193003K1.24) STEAM (CFR: 41.14): Explain the usefulness of steam tables to the control room operator	98	3.1	73
	193004	(193004K1.11) THERMODYNAMIC PROCESS (CFR: 41.14): (CONDENSERS) Describe the process of condensate depression (subcooling) and its effect on plant operation	99	2.5	74
	193009	(193009K1.07) CORE THERMAL LIMITS (CFR: 41.14): Describe factors that affect peaking and hot channel factors	100	3.3	75
	Subtotal			N/A	3
Tier 4 Point Total				N/A	**6**

Form 4.1-1 Record of Rejected Knowledge and Abilities

Refer to Examination Standard (ES)-4.2, “Developing Written Examinations,” Section B.3, for deviations from the approved written examination outline.

Tier/Group	Randomly Selected K/A	Reason for Rejection
RO 1/1 (4)	000026 G2.1.43	Cannot develop adequate question. Replaced with 2.1.19
RO 1/1 (5)	000027 AA2.13	Cannot develop adequate question. Replaced with AA2.11
RO 1/2 (19)	000024 AA2.09	Overlap with operating exam. Replaced with AA2.06
RO 1/2 (25)	WE16 EA1.04	Overlap with operating exam. Replaced with EA1.05
RO 2/1 (50)	076 A1.03	Overlap with operating exam. Replaced with A1.02
RO 2/1 (51)	076 K3.16	Invalid system combination in K/A. Replaced with K3.05
RO 2/1 (53)	103 K4.05	Cannot develop adequate question. Replaced with K4.01
--10/22/22		
RO 1/1 (8)	2.3.12	Cannot develop adequate question. Replaced with 2.2.44
RO 1/1 (17)	WE11 EK1.08	Cannot develop adequate question. Replaced with EK1.07
RO 1/2 (19)	000024 AA2.06	Cannot develop adequate question. Replaced with AA2.10
RO 1/2 (22)	000076 AK3.03	Cannot develop adequate question. Replaced with AK3.04
RO 2/1 (28)	004 A3.01	Cannot develop adequate question. Replaced with A3.08
RO 2/1 (50)	076 A1.02	Cannot develop adequate question. Replaced with A1.06
SRO 2/1 (89)	064 2.1.47	Cannot develop adequate question. Replaced with 2.1.9
SRO 3 (95)	2.1.46	Cannot develop adequate question. Replaced with 2.1.35
-11/7/22		
RO 2/1 (28)	004 A3.01	Cannot develop adequate question. Replaced with A3.06
RO 2/1 (50)	076 A1.02	Cannot develop adequate question. Replaced with A1.03
RO 2/2 (62)	072 A4.01	Adjustment function not performed at Salem. Replaced with A4.02
SRO 2/2 (93)	029 A2.01	System not currently in use. Replaced with 075 A2.02
11/28/22		
RO 1/2 (19)	000024 AA2.06	Overlap with operating exam. Replaced with AA2.12

Facility: Salem											**Exam Date:** 2/13/23-2/23/23	
1 JPM # or title	2 Type (S/P/A)	3 ALT (Y/N)	4 LOD (1–5)	5 JPM Errors							6 U/E/S	7 Explanation
				LOD	REF	IC	TSK	CUE	CS	TL		
RO-A1a	**A**	**N**	**3**								**S**	**Provide full set of completed readings for answer key-*station agreed to provide***
RO-A1b	**A**	**N**	**3**								**S**	**Initiating cue and student handout: you're vs your. Need acceptable ranges for flowrate at time, i.e. ±1 gpm, ± 0.5 minutes for individual critical steps and task standard to allow for graph interpretation. Need completed worksheet as answer key. *Station agreed***
RO-A2	**A**	**N**	**3**								**S**	**Underscore between Day and AFD in cue and student handout-*station agreed to correct***
RO-A4	**A**	**N**	**3**								**S**	**Update task standard to include successful performance of step 1.2.b of EP-SA-325-F8-*station agreed update***
SRO-A1a	**A**	**N**	**3**								**S**	**Minor typo initial conditions and handout 'has been reduce', 'record you're'-*station agreed to update***
SRO-A1b	**A**	**N**	**3**								**S**	**Minor typo, is in a refueling outage. Fatrigue rule…and Empcenter are unavailable-*station agreed to update***
SRO-A2	**A**	**N**	**3**								**S**	**Adjust times for 2VC8, 2VC12 and 2VC14 to be over 2 seconds so 2VC13 doesn't stick out. Make 2VC12 time in the evaluation required category and add that critical step to the JPM, update task standard. *Station agreed to update***
SRO-A3	**A**	**N**	**3**								**S**	**Does the applicant need the whole procedure, or just Attachment 3?***Station agreed to review*
SRO-A4	**A**	**N**	**3**								**S**	

Facility: Salem											**Exam Date:** 2/13/23-2/23/23	
1 JPM # or title	2 Type (S/P/A)	3 ALT (Y/N)	4 LOD (1–5)	5 JPM Errors							6 U/E/S	7 Explanation
				LOD	REF	IC	TSK	CUE	CS	TL		
JPM A	**S**	**Y**	**3**								**S**	
JPM B	**S**	**N/Y**	**3**								**S**	**Add procedure steps to task standard. JPM as written is not alternate path, need to correct, less than 4 alternate path JPM. *Station agreed to review; submitted JPM does not match description***
JPM C	**S**	**Y**	**3**								**S**	**Add procedure step to task standard-*station agreed***
JPM D	**S**	**Y**	**3**								**S**	**Add procedure steps/section to task standard-*station agreed***
JPM E	**S**	**N**	**3**								**S**	**Minor typo in task standard, establishes flow; step 7.2 selected sg is less than 575 psig. Add procedure reference to task standard. Add noun names to valves for examiner information. *Station agreed***
JPM F	**S**	**Y/N**	**3**								**S**	**Add procedure steps to task standard. *Station agreed***
JPM G	**S**	**Y/N**	**3**								**S**	**Should step 3.12 be broken out into sub-steps? *Station will review***
JPM H	**S**	**Y**	**3**								**S**	**Add procedure steps to task standard, typo in step 5.2.5 'pressure decrease to' *Station agreed***
JPM I	**P**	**N**	**3**								**S**	**Typo on student handout 'nitial Conditions'. Add specific steps of procedure required to successfully complete task standard. *Station agreed***
JPM J	**P**	**N**	**3**								**S**	**Add in procedure reference to task standard, completion of S1.OP-SO-RCS-0001 section 4.4-*station agreed***

Facility: Salem											**Exam Date:** 2/13/23-2/23/23	
1 JPM # or title	2 Type (S/P/A)	3 ALT (Y/N)	4 LOD (1–5)	5 JPM Errors							6 U/E/S	7 Explanation
				LOD	REF	IC	TSK	CUE	CS	TL		
JPM K	**P**	**N**	**3**								**S**	**Add step numbers to task standard. Step 5.2.11.D.6 is not critical-*station agreed to review***

Form 2.3-3 Operating Test Review Worksheet (JPMs)

Form 2.3-3 Instructions for Completing the JPM Table

1. Enter the JPM number and/or title.

2. Enter the type of JPM—(S)imulator, (P)lant, or (A)dministrative.

3. Enter (Y)es or (N)o for an Alternate Path JPM.

4. Rate the level of difficulty (LOD) of each JPM using a scale of 1–5 (easy–difficult). A JPM containing less than two critical steps, a JPM that tests solely for recall or memorization, or a JPM that involves directly looking up a single correct answer is likely LOD = 1 (too easy). Conversely, a JPM with over 30 steps or a JPM that takes more than 45 minutes to complete is likely LOD = 5 (too difficult).

5. Check the appropriate block for each JPM error type, using the following criteria:

   - LOD = 1 or 5 is unsatisfactory (U).
   - REF: The JPM lacks required references, tools, or procedures (U).
   - IC: The JPM initial conditions are missing or the JPM lacks an adequate initial cue (U).
   - CUE: The JPM lacks adequate evaluator cues to allow the applicant to complete the task, or the evaluator cues are subjective or leading (U).
   - TSK: The JPM lacks a task standard or lacks completion criteria for a task standard (U).
   - CS: The JPM contains errors in designating critical steps, or the JPM lacks an adequate performance standard for a critical step (U).
   - TL: The JPM validation times are unreasonable, or a time-critical JPM lacks a completion time (U).

6. Mark the JPM as unsatisfactory (U), satisfactory (S), or needs enhancements (E). A JPM is (U) if it has one or more (U) errors as determined in step 5. Examples of enhancements include formatting, spelling, or other minor changes.

7. Briefly describe any JPM determined to be unsatisfactory (U) or needing enhancement (E). Save initial review comments and detail subsequent comment resolution so that each exam-bound JPM is marked by a satisfactory (S) resolution on this form.

Form 2.3-3 Operating Test Review Worksheet (Scenarios)

Facility: Salem			Scenario: 1			Exam Date:	
1 Scenario Event ID/Name:	2 Scenario event errors Realism/ Credibility	Performance Standards	Verifiable Actions	Critical Task	TS	3 U/E/S	4 Explanation
1						S	Add in applicable steps/actions for boration and turbine load reduction. Station agreed to correct.
4						S	Add in more details of steps operators will take for charging pump swap. Station agreed to correct.
Facility: Salem			**Scenario: 2**			**Exam Date:**	
1						S	Need actions documented for operators for load reduction. Station agreed to correct.
3						S	Need actions taken by operators documented. Station agreed to correct.
5						S	Need actions taken by operators documented. Station agreed to correct.
CT						S	Critical task documentation needs to be updated starting on bottom of page 25 (CT1 part 2); still mention of 3 critical tasks. Typo top of page 28. Station agreed to correct.
Facility: Salem			**Scenario: 3**			**Exam Date:**	
1						S	Add in applicable steps/actions for dilution and turbine load reduction. Station agreed to correct.
Facility: Salem			**Scenario: 4**			**Exam Date:**	
1						S	Add in PO actions for steam dumps. Station agreed to correct.
2						S	This is TS only for SRO, no verifiable actions by RO or PO. Station agreed to correct.
						S	Typo page 25 greatert than. Station agreed to correct.
							Generic comment: need to script in more operator actions so examiners can validate all actions taken. Station agreed to correct.

Form 2.3-3 Instructions for Completing the Scenario Table

1. For each scenario, enter the scenario event names and descriptions.

2. Review the individual events contained in each scenario, and identify and mark event errors:

    - The scenario guide event description is not realistic/credible—unsatisfactory (U).

    - The scenario guide event description lacks adequate crew/operator performance standards—needs enhancement (E).

    - The scenario guide event description lacks verifiable actions for a credited normal event, reactivity event instrument/component malfunction, or technical specification (TS) event (or a combination of these) (U).

    - The scenario guide event description incorrectly designates an event as a critical task (i.e., a noncritical task labeled as critical or a critical task labeled as noncritical). This includes critical tasks that do not meet the critical task criteria (i.e., the critical task does not have a measurable performance standard) (U).

    - The scenario guide event description incorrectly designates entry into TS actions when not required or does not designate entry into TS actions when required (U).

3. Based on the outcome in step 2, mark the scenario event as unsatisfactory (U), satisfactory (S), or needs enhancements (E). An event is (U) if it has one or more (U) errors as determined in step 2. Examples of enhancements include formatting, spelling, or other minor changes.

4. Briefly describe any scenario event determined to be unsatisfactory (U) or needing enhancement (E). Save initial review comments and detail subsequent comment resolution so that each exam-bound scenario event is marked by a satisfactory (S) resolution on this form.

Form 2.3-5 Written Examination Review Worksheet

Q#	1. LOK (F/H)	2. LOD (1–5)	3. Psychometric Flaws					4. Job Content Flaws				5. K/A Use Flaws		6. Source (B/M/N)	7. Status (U/E/S)	8. Explanation
			Stem Focus	Cues	T/F	Cred. Dist.	Partial	Job Link	Minutia	#/Units	Logic	Q–K/A	License Level			
1	F	2												N	S	Explanation is wrong. “Two out of three channels less than or equal to 90% of design loop flow indicates degraded RCP. Since reactor power is above P-7 (10%), no automatic reactor trip will occur.“ Should state that “that above P-7 but below P-8, one RC loop low flow does not cause an auto Rx Trip.” *Station agreed to correct-corrected*
2	H	3												M	S	
3	H	3												B	S	“C” is listed as correct answer, however explanation is backwards from actual answer. Suggest in explanation use when 1CV71 is closed, then… C. Correct. When the 1CV71 is **opened**, backpressure to the seal injection lines will lower resulting in lowering RCP seal injection flows and allowing more charging flow to the non-regen HX, thereby raising charging flow to the RCS *Station agreed to correct-corrected*
4	H	3					X							N	S	There is no mention of the size of the leak in the stem. However, the answer depends on the leak size. “A” is incorrect because “A. Incorrect. Plausible because these indications will determine a Thermal Barrier leak. Incorrect in that the 2CC131 closing will also be an indication depending on the size of the leak.” Two correct answers. Also need component noun descriptions-*station states that applicants will know components without noun descriptions and adding them in would be leading. Station also agreed to review stem conditions to ensure only one correct answer.-corrected*
5	H	3												B	S	
6	F	2												N	S	
7	F	3												N	S	Consistency in answers needs to be corrected; A-C state an emergency backup source and D states the emergency backup source-*station agreed to correct-corrected*
8	F	3												N	S	Stem needs component noun descriptors; tweak wording on C and D 'only the 21'-*station agreed to add noun descriptors and add ‘the’ to C and D-corrected*
9	F	3												B	S	Possibly integrate the noun name of 2-EOP-LOPA-1-*station agreed to add note with noun name of procedure-corrected*
10	H	3												B	S	Questionable plausibility of A and B considering we are giving them seal inlet temperature. Include bearing temperatures so they have to evaluate them too? *Station agreed to validate that both seal inlet temp and bearing*

																temps are measure off the same data point. If not, likely need to add in bearing temps into stem. If so, stem can stay as-is. Corrected-station agreed to modify stem and answers to include safe shutdown seals and #1 seals
11	H	3												M	S	
12	F	2												N	S	
13	H	3												N	S	
14	H	3												N	S	
15	H	3												M	S	In the parent question, there is mention of ring bus and voltage regulator status. Is this needed for the modified question? Answer explanation for D says Unit 2 tripped. *Station agreed to add in voltage regulator status and correct answer explanation-corrected*
16	H	3												N	S	Minor typo in answers B/C explanation clod vs cold-*station agreed to correct-corrected*
17	H	3												N	S	Stem does not mention status of RCP's. The question asks if head voiding is expected. The reference material states head voiding is expected if no RCPs are running. This implies no voiding if they are running. Therefore, not enough info to determine if head voiding is expected. Recommend add to stem no RCPs are running. Autonumbering of listed statements in stem needs to be corrected-*station agreed to add in that RCPs are not running and correct autonumbering issue-corrected*
18	H	3												N	S	Does the 22MS167 need a noun name added? *Station agreed to add noun description-corrected*
19	F	3												B	S	
20	F	2												N	S	Question is asking applicant to pick the best answer between two correct answers. A is a second correct answer. Need to make the distractor incorrect. *Station agreed to review and update to only have one correct answer-corrected*
21	H	4												N	S	
22	H	3												B	S	
23	F	2										X		N	S	Only asks what an eop symbol is. Limited testing of K/A. K/A typo listed as 2.4.29, should be 2.4.19-*station agreed to correct K/A number and update K/A match justification-corrected*
24	F	3												B	S	

25	H	3												N	S	
26	F	3					X							N	S	Question is nebulous. Asks how many RCPs are started. C could be a second correct answer. Explanation states 2 could be started if 23 RCP not available. Nothing in stem about which pump or availability. Although power is restored, could be not available for other reason. *Station agreed to add 'in order of priority' to preclude two correct answers-corrected*
27	H	3												B	S	Needs procedure reference, IAW S2.OP-AB.RCP-0001, possible exclusionary language, i.e. maximum amount of time allowed-*station agreed to add procedure reference and change stem to reflect maximum time allowed-corrected*
28	H	3												B	S	Minor typo in stem, two commas after 130; wrong importance rating, should be 3.6 not 3.8-*station agreed to correct both-corrected*
29	H	3												N	S	Answer balancing needed, possible to match 2 answers with damage to charging pump and 2 without? *Station agreed to modify D such that 2 answers would have damage to charging pump, 2 without-corrected*
30	H	3												M	S	Add valve noun names-*station agreed to add-corrected*
31	F	2												B	S	
32	F	2												N	S	
33	H	3												N	S	
34	H	3												N	S	Minor typo in K/A match statement, PORC vs PORV-*station agreed to correct-corrected*
35	H	3												B	S	Need to correct the "C" answer explanation to 2/4.*station agreed to correct-corrected*
36	F	3												B	S	
37	H	3												B	S	"Safeguards" applies to all SEC MODES? Wording, typically ask SEC MODE?-*station states stem conditions require determining SEC mode to determine correct answer*
38	H	3												M	S	
39	F	3												B	S	
40	H	3												N	S	
41	H	3												N	S	First bullet in stem should say main steam isolation "occurred" due …-*corrected*

42	F	3												B	S	
43	H	3												B	S	Needs procedure reference for K/A match-station agreed to review/correct
44	F	3												N	S	
45	F/H	3												N	S	
46	F	3												N	S	Change ground reading to require candidates to recall ground threshold. 0k is too obvious. Answer wording for B and C needs adjustment to more accurately reflect the sequence of events. Open, evaluate grounds, if no change, reclose, etc…*station agreed to change all answer second parts for balance-corrected, changed to 40k and balanced answers*
47	F	2												B	S	
48	F	3												N	S	
49	F	3												N	S	
50	H	3												N	S	Remove 'than before' from stem so all answers make sense. *Station agreed to correct.* The explanation for distractor "A" second part needs to say correct-station agreed to correct-*corrected*
51	H	3												N	S	Correct answer explanation needs to be fixed to match stem conditions. The explanation for B is backwards. The air line fails therefore the valve fails open. "21SW122 fails open on loss of air (fails closed on loss of 125VDC power). When the 21SW122 closes, SW flow will lower through the CCW HX causing CCW temperature to rise and therefore the outlet temperature of 21RHR HX to rise and lowering the cooldown rate." *Station agreed to correct-corrected*
52	F	3												M	S	Only should not be capitalized, *station agreed to correct-corrected*
53	F	3												N	S	
54	F	3				X								N	S	A and B are not plausible. The wording of the question leads to knowing the answer has to be .3. If 1.5 was correct then .3 would be correct and there would be 2 correct answers. *Station agreed to add qualifier in step 1 for 'exceeds a minimum of; move TS 3.6.1.4 statement-corrected*
55	F	3												M	S	

56	H	3												N	S	Need procedure reference for K/A match. Control room log? *Station agreed to correct-corrected*
57	H	3				X								B	S	A and D are not plausible. If 3300 was correct then 2300 is also correct. *Station agreed to change stem conditions to ask the design temperature of the CETs-corrected*
58	H	3												B	S	Change bounding to binding-*station agreed to correct-corrected*
59	H	3												B	S	
60	H	3												N	S	Minor typo in B answer explanation; systen vs system-*station agreed to correct.* Stem (Then bullet) needs to say waterboxes – plural.-station agreed to correct-*corrected*
61	H	3												N	S	
62	H	3												N	S	
63	F	3												M	S	
64	F	3												B	S	Add noun description to 2CN47-*station agreed to add-corrected*
65	F	3												N	S	
66	F	3												N	S	
67	F	3												B	S	Q67 is a tier 3 question. Listed as tier 4. Correct on second page of question. *Station agreed to correct-corrected*
68	H	3												B	S	
69	F	2												B	S	
70	F	2												B	S	The question is backwards from an operations view. To maintain reactor power and Tavg stable over the next hour, the operator must add __(1)__ reactivity because the xenon-135 concentration will be __(2)__? Suggest: Xenon-135 concentration will be __(1)____? To maintain reactor power and Tavg stable over the next hour, the operator must add __(2)__ reactivity. *Station agreed to correct logic-corrected*
71	F	3												B	S	Answer a is shifted to the right-*station agreed to correct-corrected*
72	F	3												N	S	

73	H	3												M	S	MS10s need noun names-*station agreed to add-corrected*
74	H	3												B	S	
75	H	3												B	S	
76	H	3												M	S	Question doesn't seem realistic. The reactor tripped with a partially open PORV. Pressure will go down and stabilize at some point. Not sure what time period but is the operator supposed to know that it takes longer than 5 minutes. How did you get to 500 psig. Pressurizer empties and you end up at the pressure that is saturation for the hot leg. Need to rethink this and make it thermodynamically correct. *Station agreed to modify-station replaced question.*
77	F	3												B	S	B doesn't seem plausible. They state you have entered the FRP, then they say do the step in loca 1 because you aren't supposed to do FRP's while in loca 1. Not logical. Easy to throw out. *Station agreed to remove statement in stem that the crew entered FRTS-1.* Stem: Then first bullet should be plural – "exists"-station agreed to correct-*corrected. Following validation, adjusted answer to remove superseded and changed to 'transition to'*
78	H	3												N	S	The answer is "Determine if a Unit shutdown is required in preparations for removing 21 RCP from service." However what step 3.10 says is "If unit shutdown is determined, then…" Don't think they mean the same thing. Need a more definitive action step to test. Also stem question asks what will the CRS direct. Determining if shutdown is required is not something he is going to direct to the RO's. Change motor flange vibrations to be closer to threshold so applicants must discern-*station agreed to raise vibration in stem to be accurate for AB entry, and change answer B to be an operator direction-corrected*
79	H	3												N	S	
80	H	3												N	S	
81	H	3												B	S	Direct lookup. I don't need any knowledge to read the stem and then decide there is one condensate pump running and 0 heater drain pumps. The second half of the question is a should in the procedure, Is the person incorrect to use the normal procedure? Under references provided, it says Attachment 3 of S2.OP-AB.CN-0001, but attached reference is Attachment 2-*station evaluating resampling of K/A-station replaced question with new question, replacing the original proposed bank question*
82	H	3												N	S	Not SRO. Can the question be answered solely with system knowledge. What indication is the IRM setpoint for P-6 is system knowledge and whether the channel is

																going to process a reactor trip on loss of instrument power is system knowledge. If a person knows the reactor did not trip they are not going to pick the reactor trip procedure. Therefore they will choose the NI abnormal-*station modified question to include TS bases for 24 hour action time. Post validation, added wording to stem conditions a valid P6 light is now lit on 2RP4*
83	H	3												N	S	Correct answer explanation needs more info. Steam line differential pressure (SI?) Importance should be 4.3 not 4.6-*station agreed to edit correct answer explanation and importance rating-corrected*
84	H	3				X								M	S	A and C do not appear plausible based on procedure title. Original question seems more plausible-*station agreed to change A and C to be more in line with bank question, adjusting part 2 in question as necessary.* Dump steam at "maximum" rate? How does that equate with 100°F/HR? Looking at potential future appeal? *Station states that the wording selected is directly out of the procedure. Station rewrote second part of question to evaluate procedure basis for starting of RCPs-corrected. Post validation-added 1200F CET(needed info for EOP step) and adverse containment to change correct answer to A.*
85	H	3												N	S	Cannot ask best answer. The incorrect answer has to be wrong. Stem possibly cues correct answer. Change 'dump steam' to 'reduce 21 SG pressure'-*station agreed to modify stem conditions to remove BEST reference, add in procedure reference for direction, and change 'dump steam' to 'reduce 21 SG pressure'-corrected; station changed stem question and changed dump steam to reduce 21 SG pressure*
86	H	3												M	S	
87	H	4												B	S	Is PZR Level at 15% and lowering enough? Maybe rapidly lowering? *Station stated that in TRIP 4 cooldown rate is restricted, so cooldown could not potentially be cause of rapid level reduction with 2CV55 fully open, will change pzr level to 10%-corrected*
88	H	3												N	S	
89	F	1												N	S	LOD 1, too simple-*station agreed to review. Station replaced with Braidwood bank question*
90	H	3												M	S	
91	H	3												N	S	The difference between C and D is the slang term "enter TS." All TS are applicable at all times and when the LCO is not met then the action statement is required to be performed within the time specified. Since "enter TS" is not defined both answers C and D are correct. TS 3.5.2 is mentioned in supplied references; it is not needed-*station agreed to review wording-corrected*
92	H	3												B	S	Not SRO. Can the question be answered solely with system knowledge? The first part about the failed transmitter and what direction level will fail is system knowledge. The stem states the channel 1 was placed in

																the tripped condition. It is RPS system knowledge that 2/3 Pzr levels will trip the reactor. Therefore the question can be answered with only system knowledge.-*station agreed to evaluate/review-**station replaced stem with reportability question***
93	H	4												N	S	Page 69, reference does not belong with this question-*station agreed to remove-corrected*
94	F	3												N	S	Minor typo in correct answer explanation and K/A match explanation-*station agreed to correct-corrected*
95	H	3												N	S	
96	F	3												M	S	Typos in stem ‘conditions results’ Protected Equipment Program missing first quote. Typo in correct answer explanation F:EX-*station agreed to correct.* Unsure of wording in answer choice 1-*station stated that fire in a(4) is commonly used phrase referring to risk mitigation*
97	H	3												N	S	
98	F	3												B	S	
99	H	3												B	S	
100	H	3												B	S	

Form 2.3-5 Instructions for Written Examination Review Worksheet

Refer to ES-4.2 for the definitions of terms used in this worksheet for the written examination. Review each question (Q) as submitted and as subsequently revised and document the following in the associated worksheet columns:

1. Enter the level of knowledge (LOK) as either (F)undamental or (H)igher cognitive level.
2. Enter the level of difficulty (LOD) from 1 (easy) to 5 (difficult); mark direct lookup questions (applicant can directly determine the answer from the provided reference) as LOD 1. A question is (U)nsatisfactory if it is LOD 1 or LOD 5.
3. Check the appropriate box if a psychometric flaw is identified:
   - "Stem Focus": The stem lacks enough focus to elicit the correct answer (e.g., unclear intent, more information is needed, or too much needless information). This is an (U)nsatisfactory question.
   - "Cues": The stem or one or more answer choices contains cues (e.g., clues, specific determiners, phrasing, length). This is an (U)nsatisfactory question.
   - "T/F": All of the answer choices are a collection of unrelated true/false statements. This is an (U)nsatisfactory question.
   - "Cred. Dist.": The distractors are not credible; single implausible distractors require (E)nhancement, and more than one noncredible distractor in the same question results in an (U)satisfactory question.
   - "Partial": One or more distractors are partially correct (e.g., if the applicant can make unstated assumptions that are not contradicted by the stem). This is an (U)nsatisfactory question.

4. Check the appropriate box if a job content flaw is identified:
   - "Job Link": The question is not linked to the job requirements (i.e., the question has a valid knowledge or ability (K/A) but, as written, is not operational in content). This is an (U)nsatisfactory question.
   - "Minutia": The question requires the recall of knowledge that is too specific for the closed-reference test mode (i.e., it is not required to be known from memory). This is an (U)nsatisfactory question.
   - "#/Units": The question contains data with an unrealistic level of accuracy or inconsistent units (e.g., panel meter in percent with question in gallons). This is an (U)nsatisfactory question.
   - "Logic": The question requires backward or reverse logic or application compared to the job requirements. This is an (U)nsatisfactory question.
5. Check the first box if a K/A mismatch flaw exists. Check the second box if the question is flawed because it is written at the wrong license level. Either condition results in an (U)nsatisfactory question.
6. Enter the question's source: (B)ank, (M)odified, or (N)ew. Verify that (M)odified questions meet the criteria of ES-4.2.
7. Based on the review performed in steps 2–5, mark the question as (U)nsatisfactory, in need of (E)nhancement, or (S)atisfactory.
8. Fully explain the reason for any (U) in column 7 (e.g., how the psychometric attributes are not being met).
9. Save the initial review comments and detail subsequent comment resolution so that each exam-bound question is marked by an (S) on this form.

RO EXAMINATION ANSWER KEY

SALEM 2023 NRC Exam (21-01 ILOT) **– (Rev. 0)**

1 **10-150** **Points: 1.00**

Given:

- Salem Unit 2 is at 25% reactor power.
- The RO reports the following flow indications on 21 RC loop:
  - Channel 1 – 89%
  - Channel 2 – 89%
  - Channel 3 – 92%

Which of the following describes the expected response?

A. Per the Alarm Response Procedure, manually trip the reactor, stop 21 RCP, and enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

B. RC flows are degraded but above the RPS setpoint for low RCS flow; power operation may continue.

C. Automatic reactor trip based on 2/3 low flow channels on one RC loop with power greater than P-7; perform immediate actions per 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

D. Automatic reactor trip based on 1/3 low flow channels on one RC loop with power greater than P-7; perform immediate actions per 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

Answer: A

Explanation / Justification:

A. Correct. Two out of three channels less than or equal to 90% of design loop flow indicates degraded RCP. Since reactor power is above P-7 (10%), no automatic reactor trip will occur. However, per S2.OP-AR.ZZ-0004 for OHA D-28, RCP BKR OPEN/FLO LO, with a degraded RCP flow, the RCP must be stopped, the reactor manually tripped, and enter 2-EOP-TRIP-1 since Tech Specs does not allow for three loop operation in Mode 1.

B. Incorrect. Plausible if the operator fails to recall the RC low flow setpoint of less than or equal to 90% design flow and believes the RCP is only degraded and does not meet the requirements for stopping the RCP. Incorrect in that the RCP is below the 90% flow requirements and must be stopped and the reactor tripped.

C. Incorrect. Plausible if the operator believes that the conditions meets the automatic RPS reactor trip for low RC flow. Incorrect in that above P-7 but below P-8, one RC loop low flow does not cause an auto Rx Trip.

D. Incorrect. Plausible if the operator believes that the conditions meets the automatic RPS reactor trip for low RC flow. Incorrect in that above P-7 but below P-8, one RC loop low flow does not cause an auto Rx Trip.

Technical References:	Logic drawing 221054, simplified ESF-2, S2.OP-AR.ZZ-0004 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	10-Reactor Trip
K/A #:	EPE 7 EA2.08 Ability to determine and/or interpret the following as they apply to a Reactor Trip: RCS loop flow rates
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the operator must assess and/or interpret the RC loop flow conditions, recall RPS trip logic for RC loop low flows, and determine how the plant or the operators should respond.

Additional Reference: OHA Window D-28

S2.S2.OP-AR.ZZ-0004(Q)

D-28 (continued)

3.1 **DETERMINE** cause of alarm from Annunciator CRT.

CRT POINT	DESCRIPTION
413	Reactor Coolant Loop 21 Low Flow Channel I
414	Reactor Coolant Loop 21 Low Flow Channel II
415	Reactor Coolant Loop 21 Low Flow Channel III
416	RCP 21 Breaker Open Channel I

3.2 IF cause of alarm is failed instrument,

THEN NOTIFY Maintenance Controls to remove failed instrument from service IAW S2.OP-SO.RPS-0007(Q), RCP Channel Trip/Restoration,

AND REFER to Technical Specifications for applicability.

3.3 IF alarm is due to intentionally stopping 21 RCP as part of normal plant operations,

THEN NO further action is required.

3.4 IF a Reactor trip occurs,

THEN GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

3.5 IF Reactor Coolant System flow degradation exists,

THEN:

A. **TRIP** the Reactor.

B. **STOP** 21 RCP.

C. IF RCP shutdown was due to RCP Seal Leakoff flow greater than or equal to 6 gpm, THEN simultaneously **PERFORM** the following:

- Between 3-5 minutes after stopping 21 RCP, **CLOSE** 21CV104, SEAL LEAKOFF.
- GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

D. GO TO 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

2 **10-151** **Points: 1.00**

Given:

- A small break LOCA and a Loss of ALL Offsite Power occurred on Salem Unit 2

Current plant conditions are as follows:

- SG pressures are 1015 psig and stable
- RCS pressure is 1600 psig and slowly lowering
- Highest CET temperature is 590 °F and lowering
- RCS T-Hot temperatures are 554 °F and lowering
- RCS T-Cold temperatures are 548 °F and stable
- 2B 4KV Vital Bus is locked out on Bus Differential

What is the primary method for RCS heat removal for current plant conditions?

A. Forced RCS Cooling

B. Natural Circulation

C. Reflux Boiling

D. ECCS Injection and Break flow

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may forget that a LOOP has occurred and the 4KV group busses are not energized to run the RCPs for forced flow cooling.

B. Correct. Forced cooling is not available due to Loss of All Offsite Power. Only one Charging Pump is running and no SI pumps based on RCS pressure above the shutoff head (1540 psig), therefore, ECCS injection flow and break flow are insufficient for RCS heat removal. Natural circulation conditions are met based on the following parameters: (1) subcooling > 0°F. RCS pressure at 1600 psig has a saturation temperature or approx. 606 °F. Hottest CET is 590 °F, therefore subcooling is approx. 606 – 590 = 16 °F of subcooling, (2) SG pressures are stable and lowering, (3) RCS T-Hots are stable or dropping, (4) CETs are lowering, and (5) RCS T-colds are at saturation temp for SG pressure. SG pressure is 1015 psig (pre-set pressure for MS110s) which is a saturated temp from the outlet of the SGs of approx. 548 °F. Therefore per 2-EOP-LOCA-2 step 19.1 conditions for Natural Circulation are met and will be the RCS heat removal method.

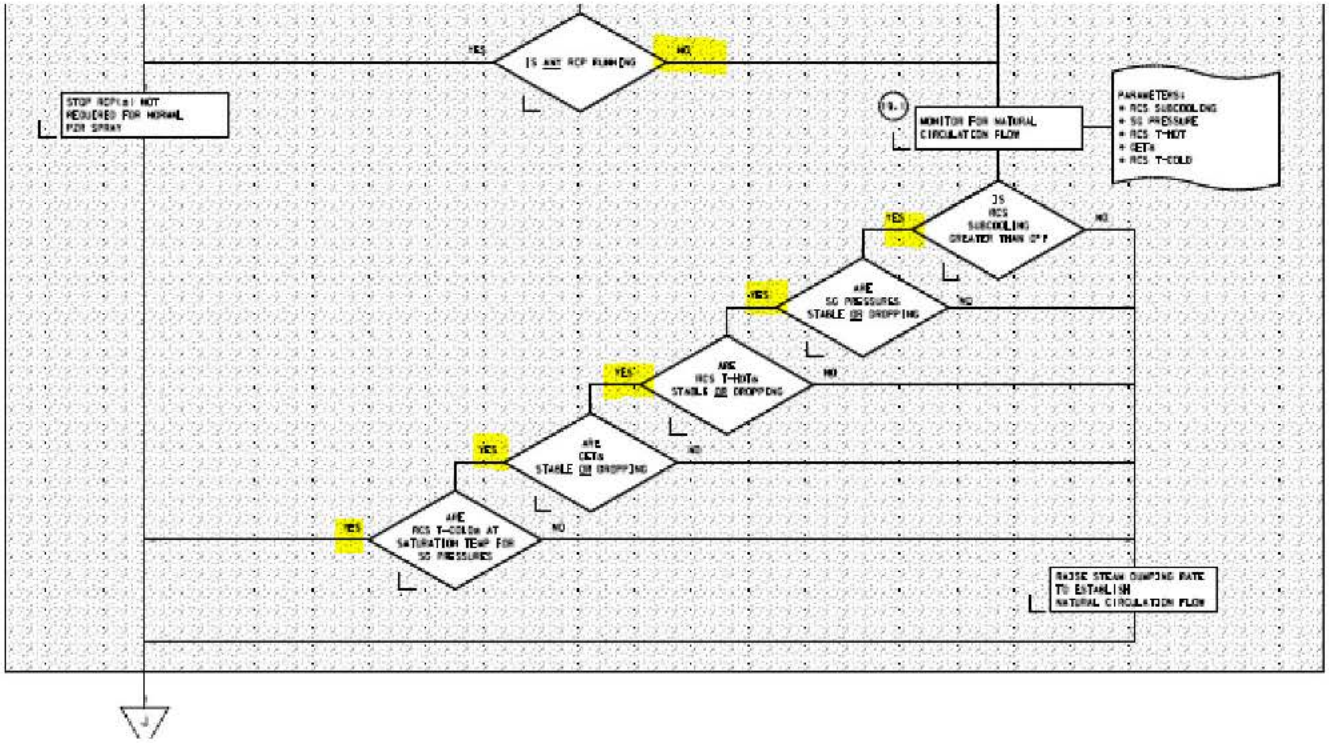
C. Incorrect. Plausible because Reflux cooling is a method for RCS cooling if none of the other methods are available.

D. Incorrect. Plausible because during some LOCA events, ECCS injection and break flow is the primary method for RCS heat removal (e.g., LBLOCA).

Technical References:	2-EOP-LOCA-2 (R42), ASME Steam Tables (year 2000)
Proposed References to be provided:	**ASME Steam Tables Rev. 2000**
Learning Objective:	NOS05LOCA02-06, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in the EOP for POST LOCA COOLDOWN AND DEPRESSURIZATION
Question Source:	Modified – Callaway 2017 NRC RO8
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10-Small Break LOCA
K/A #:	EPE 9 EK1.03 Knowledge of the operational implications and/or cause and effect relationship of the following concepts as they apply to Small Break LOCA: RCS heat removal
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant must assess plant conditions and determine the method of RCS heat removal during a small break LOCA.

Additional Reference:



Parent Question: Callaway 2017 NRC RO8

Modified question to change the answer to Natural Circulation is the RCS heat removal method.

Question # 8

A loss of off-site power and a small break LOCA occurred.

Current plant conditions are as follows:

- RCS pressure is 1800 psig and slowly lowering
  - SG pressures are 1135 psig and stable
- Highest Core Exit Thermocouple is 630°F and stable
- RVLIS Pumps Off indicates 60%
- RCS Hot Leg Temperatures are 562°F and stable
- RCS Cold Loop Temperatures are 530°F and stable
- NB01 is locked out
- 'B' CCP, tripped due to instantaneous overcurrent

What is the RCS Heat removal process(es) for the current plant conditions?

A. Forced RCS Circulation

B. Natural Circulation

C. Reflux Boiling

D. ECCS Injection Flow AND Small Break LOCA flow

Answer: C

3 **11-200** **Points: 1.00**

Given:

- Salem Unit 1 is experiencing a RCS leak inside containment.
- The crew is performing S1.OP-AB.RC-0001, Reactor Coolant System Leak.
- 11 charging pump is in service.
- The RO is attempting to stabilize PZR level by adjusting charging flow using the 1CV55, CHARGING FLOW CONTROL VALVE, and 1CV71, CHG HDR PCV.

Which of the following completes the below statement?

When 1CV71 is throttled…

A. open, seal injection rises and charging flow rises.

B. open, seal injection flow lowers and charging flow lowers.

C. closed, seal injection flow rises and charging flow lowers.

D. closed, seal injection lowers and charging flow rises.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because opening the 1CV71 will result in charging flow rising. Incorrect in that 1CV71 acts as backpressure control valve that affects both charging and seal injection flows. Opening the 1CV71 will cause backpressure in the charging line to lower resulting in an increase in charging flow but lower flow to the RCP seal injection lines.

B. Incorrect. Plausible because opening the 1CV71 will result in seal injection flow lowering. Incorrect in that 1CV71 acts as backpressure control valve that affects both charging and seal injection flows. Opening the 1CV71 will cause backpressure in the charging line to lower resulting in an increase in charging flow but lower flow to the RCP seal injection lines.

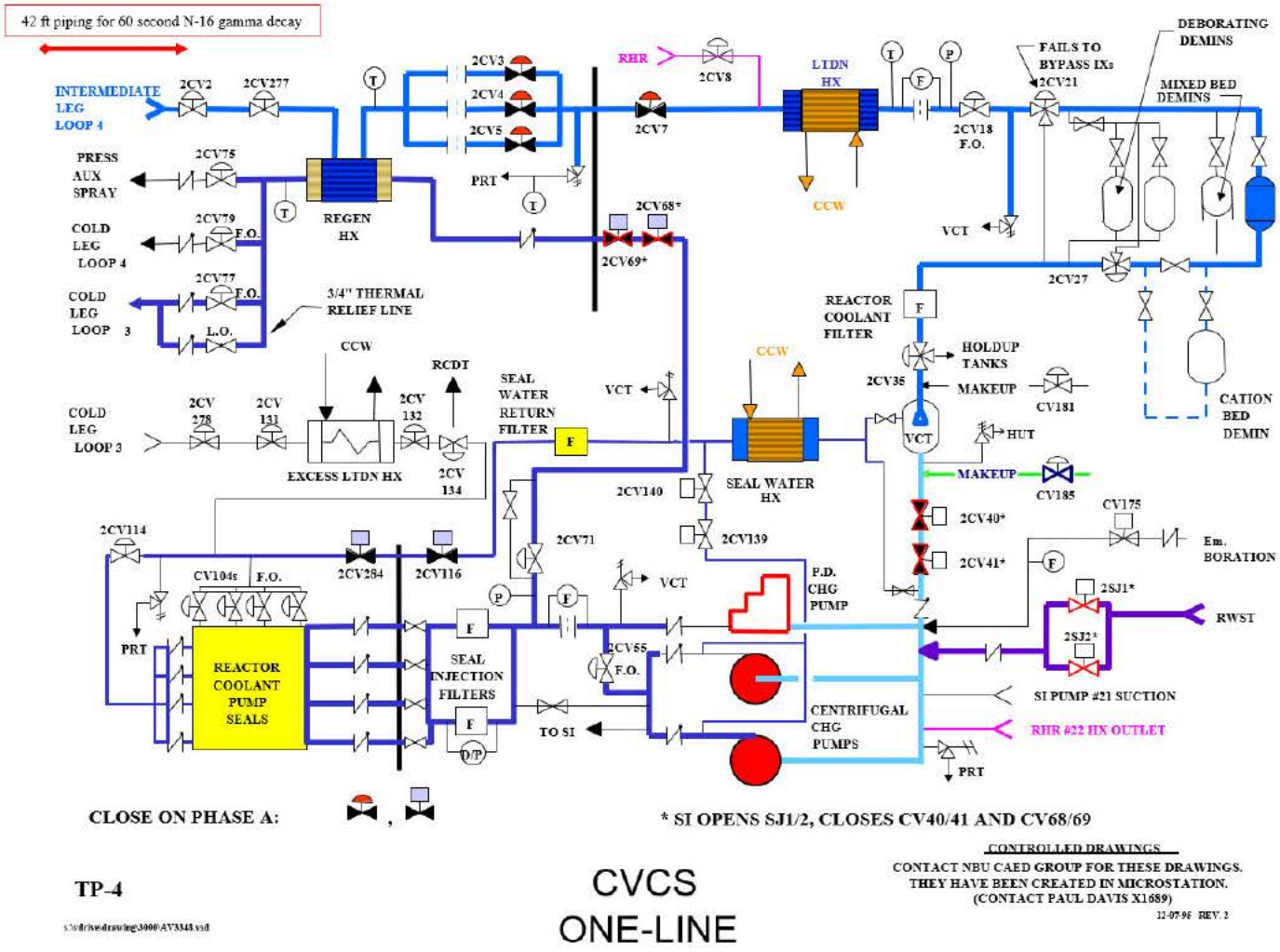
C. Correct. When the 1CV71 is opened, backpressure to the seal injection lines will lower resulting in lowering RCP seal injection flows and allowing more charging flow to the non-regen HX, thereby raising charging flow to the RCS.

D. Incorrect. Plausible if the operator incorrectly recalls the location and operation of the 1CV55 and 1CV71 in the CVCS system and believes that the 1CV71 supplies flow the RCP seal injection only, thus if the 1CV71 is closed more flow will be diverted to charging flow and less to seal injection. . Incorrect in that 1CV71 open will cause seal injection to lower and charging flow to rise.

Technical References:	S1.OP-AB.RC-0001 (R18), CVCS one-line dwg
Proposed References to be provided:	None
Learning Objective:	NOS05CVCS00-18, Objective 4. Describe the function and how their normal and abnormal operations affects the Chemical and Volume Control System: Seal Pressure Control Valve, CV71.
Question Source:	Bank – Bryon 2017 NRC RO4
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	11-Loss of Reactor Coolant Makeup
K/A #:	APE 22 AA1.05 Ability to operate and/or monitor the following as they apply to Loss of Reactor Coolant Makeup: RCP seal backpressure regulator valves.
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator how operating the 1CV71 will effect charging flow and RCP seal injection flow during response to an RCS leak to stabilize PZR level.

Additional Reference: CVCS one-line



4 **11-201** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

The operating crew suspects a RCP Thermal Barrier leak.

Which of the following indications would provide corroborating evidence for this condition, consider each choice separately?

1. 2R17A/B in Warning
2. PZR level rising
3. CCW Surge Tank level rising
4. Charging flow lowering
5. 2CC131 goes closed

A. 1 and 3 Only

B. 2 and 4 Only

C. 2, 4, and 5 Only

D. 1, 3, and 5 Only

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because these indications will determine a Thermal Barrier leak. Incorrect in that the 2CC131 closing will also be an indication depending on the size of the leak.

B. Incorrect. Plausible because these would be indications if the operator believes that a Thermal Barrier leak will cause CCW to leak into the RCS system.

C. Incorrect. Plausible because these would be indications if the operator believes that a Thermal Barrier leak will cause CCW to leak into the RCS system.

D. Correct. A Thermal Barrier leak will cause CCW surge tank to rise as RCS will be leaking into the CCW system, RCS in-leakage will also be identified as a rise in the 2R17A or B radiation monitors that monitors activity in the CCW system, and the 2CC131 valve will also close on high CCW flow to the Thermal Barrier due to a thermal barrier leak, Therefore, choices 1, 3, and 5 are the correct indications for a Thermal Barrier leak into the CCW system.

Technical References:	S2.OP-AB.CC-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05CCW000-16, Objective 4.c. Describe how the following components impact the Component Cooling Water System during normal and abnormal conditions:
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	11-Loss of Component Cooling Water / Conduct of Operations
K/A #:	APE 26 G2.1.19 Ability to use available indications to evaluate system or component status.
Importance:	3.9
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being ask to determine which indications in the control room will corroborate a thermal barrier leak in the CCW system.

5 **11-202** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- PZR Pressure controlling channel fails HIGH

Which of the following completes both statements concerning the effect from this failure?

Based on the conditions above, the __(1)__ reactor trip setpoint on the **unaffected** RCS loops will __(2)__.

A. (1) OTΔT (2) lower

B. (1) OTΔT (2) rise

C. (1) OPΔT (2) rise

D. (1) OPΔT (2) lower

Answer: A

Explanation / Justification:

A. Correct. The failed PZR pressure controlling channel will result in PZR pressure lowering. The lowering PZR pressure is one input into OTDT to determine its trip setpoint. As RCS pressure lowers, the OTDT setpoint will lower. With no operator action, the PZR pressure will continue to rapidly lower until the OTDT reactor trip setpoint is reached first (per S2.OP-AB.PZR-0001 bases section for Attachment 1).

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the Low PZR Pressure Rx Trip is rate compensated as pressure lowers and will cause the Rx trip to occur at a higher pressure. Incorrect in that test runs in the simulator determined that OTDT will trip the reactor first.

C. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse how PZR pressure affects the OTDT setpoint. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse how PZR pressure affects the OTDT setpoint. Part 2 is incorrect. Plausible because the Low PZR Pressure Rx Trip is rate compensated as pressure lowers and will cause the Rx trip to occur at a higher pressure. Incorrect in that test runs in the simulator determined that OTDT will trip the reactor first

Technical References:	S2.OP-AB.PZR-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05ABPZR1-05, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.PZR-0001 and the bases for the actions.
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.5 / 45.13
Level:	RO
Category/System:	11-Pressurizer Pressure Control System
K/A #:	APE 27 AA2.11 Ability to determine and/or interpret the following as they apply to a PZR pressure Control System Malfunction: RCS pressure
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking to interpret a failed high PZR Pressure controlling channel and the effects on RCS pressure. The applicant needs to determine that the failure will cause RCS pressure to lower which will result in the OTDT setpoint to lower. The applicant is also asked which reactor trip will first based on lowering RCS pressure.

6 **10-152** **Points: 1.00**

Given:

- Salem Unit 2 is experiencing a Steam Generator Tube Rupture
- The crew is performing actions per 2-EOP-SGTR-1, Steam Generator Tube Rupture
- The CRS, RO, and BOP operators are the only licensed operators in the control room
- The RO needs to go to the Control Equipment Room (back rack area) to perform an action

Per OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, which of the following describes the requirements for this activity?

The RO…

A. must be relieved by a licensed operator other than the on-shift Unit 2 BOP operator.

B. may leave as long as the on-shift Unit 2 BOP operator remains “At the Controls” area.

C. may leave without being relieved by another licensed operator with CRS permission ONLY.

D. must be relieved by another licensed operator and a full turnover must be performed.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that during accident conditions two licensed operators must be in the control room or ATC at all times. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

B. Correct. Per OP-AA-101-111, at least one RO must be in the "At The Controls" (ATC) area (or inner-horseshoe) at all times during Modes 1-6, except for the time period required to acknowledge the receipt of an alarm outside the this area or to perform actions required for safe plant operation.

C. Incorrect. Plausible because the operator may believe SRO permission is required to leave the inner-horseshoe area. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

D. Incorrect. Plausible because the operator may believe that during accident conditions two licensed operators must be in the control room or ATC at all times. Incorrect in that only one licensed RO is required to be in the ATC or inner-horseshoe during Modes 1-6.

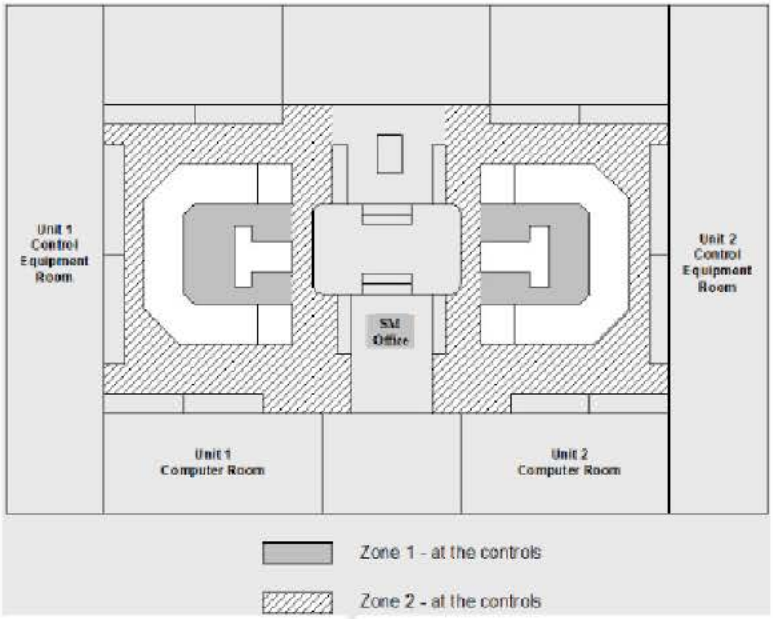
Technical References:	OP-AA-101-111 (R11)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-15, Objective 2.f and 2.g. Describe the position and purpose of the following positions in accordance with OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, and OP-AA-101-112, Roles and Responsibilities of Off-Shift Personnel: Reactor Operator (RO) and Plant Operator (PO)
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.13
Level:	RO
Category/System:	10-Steam Generator Tube Rupture / Conduct of Operations
K/A #:	EPE 38 G2.1.3 Knowledge of shift or short-term relief turnover practices.
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator knowledge of short-term reliefs during an accident conditions.

Additional Reference: OP-AA-101-111

ROLES AND RESPONSIBILITIES OF ON SHIFT PERSONNEL **Rev: 11**

Attachment 4, Salem "At the Controls Area



Whenever the on-duty SM leaves the Control Area, the Unit 2 CRS assumes Control Room Command and Control unless the SM designates a different SRO.

- At least one RO is to be within Zone 1 at all times except for the time period required to acknowledge the receipt of an alarm outside this area or to perform actions required for safe plant operation.
  (Modes 1 - 6)
- SRO with command and control function may not enter the computer or equipment rooms.
- The on-duty SM and CRS(s) shall remain within the Protected Area.
- The STA shall be within 10 minutes of the Control Room Area in Modes 1 - 4.
- The STA shall be within 30 minutes of the Control Room Area in Modes 5 - 6, Defueled.

Modes 1 - 4 (per unit)

- SRO shall remain within the Control Room area. This SRO may not go in the opposite unit's computer and equipment rooms.

7 **10-153** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A reactor trip and Safety Injection occur due to the rapid depressurization of all steam generators.
- 2H and 2E 4KV Group Busses fail to auto transfer and are now de-energized
- The crew is performing actions in 2-EOP-LOSC-2, Uncontrolled Depressurization of All Steam Generators, to restore the 4KV Group Busses

Which of the following describes the current status of the PZR Backup Heaters?

A. ONLY 21 Backup Group is available, 22 Backup Group can be locally transferred to an emergency backup source.

B. ONLY 22 Backup Group is available, 21 Backup Group can be locally transferred to an emergency backup source.

C. NEITHER 21 or 22 Backup Groups are available, Backup Groups can be locally transferred to an emergency backup source.

D. BOTH 21 and 22 Backup Groups are available, Backup Groups will automatically transfer to the emergency backup source.

Answer: A

Explanation / Justification:

A. Correct. The preferred or normal power source for the PZR Heater Backup Groups is; Group 1 (21) – 2G and Group 2 (22) – 2E 4KV Group Busses. The PZR Heaters can be manually transferred to the emergency backup source; Group1 (21) from 2C and Group 2 (22) from 2A 460 V vital bus. EOP-LOSC-2 step 26 perform actions to restore the normal group busses or transfer PZR heater groups to the emergency backup power.

B. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters.

C. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters and believe that both PZR heaters are unavailable. Incorrect in that with 2H and 2E unavailable, PZR Heater Group 1 is available from 2G Group Bus.

D. Incorrect. Plausible because the operator may incorrectly recall the normal power sources for the PZR Heaters and believe that both PZR heaters are available. Incorrect in that with 2H and 2E unavailable, only PZR Heater Group 1 is available from 2G Group Bus. Incorrect in that the Heater Groups must be manually transferred to the vital 460V busses.

Technical References:	2-EOP-LOSC-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPL-12, Objective 5.b. State the power supply to the following Pressurizer Pressure and Level Control System components: Backup Heaters
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	10-Steam Line Rupture
K/A #:	W E12 EK2.03 Knowledge of the relationship between Uncontrolled Depressurization of all Steam Generators and the following systems or components: AC electrical distribution system
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant about the normal AC power source to the PZR Heater backup groups and the availability of the emergency AC backup power source. EOP-LOSC-2 step 26 looks to restore the normal power source for the PZR heaters or transfer to the emergency backup source. Normal power is preferred to reduce the load on the EDGs,

8 **11-203** **Points: 1.00**

Given:

- Unit 2 is at 100% Power

Then:

- OHA G-22, FW HTR IN VLV TRIP & LVL HI, is Alarming
- PO reports high level in 21A Low Pressure Feedwater Heater

Which of the following describes the effect on plant operation?

1. Based on the above condition, __(1)__ CN22(s) will be closed.
2. Actual reactor thermal power will be __(2)__.

A. (1) ALL (2) higher

B. (1) ALL (2) lower

C. (1) Only 21 (2) lower

D. (1) Only 21 (2) higher

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that any high level in one of the LP heater strings will result in isolation of all LP feedwater heater strings.

B. Incorrect. Plausible because the operator may incorrectly interpret how the isolation of the FW heater string will affect FW temperatures to the reactor and how the FW temperature change will affect reactor power and NIS indications.

C. Incorrect. Plausible because the operator may incorrectly interpret how the isolation of the FW heater string will affect FW temperatures to the reactor and how the FW temperature change will affect reactor power and NIS indications.

D. Correct. High water level in 21A Low Pressure feedwater heater will only close the associated 21CN22 inlet valve. Based on one LP FW heater string out of service, colder feedwater will enter the reactor. The cooler FW will have the effect of raising reactor thermalmpower and also provide some shielding to the PR NIS and result in the actual NI reactor power being lower than actual. As a result reactor power should be determined based on RCS ΔTs and turbine load should be reduced to prevent exceeding reactor thermal power limits.

Technical References:	S2.OP-AR.ZZ-0007 (R67)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCN01-07, Objective 3. Describe, in general terms, the actions taken in AB.CN-0001 and the bases for the actions in accordance with the technical bases document.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.12
Level:	RO
Category/System:	11-Loss of Main Feedwater / Equipment Control
K/A #:	APE 54 G2.2.44 Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is asked how the high level in a feedwater heater will affect the isolation of the three LP feedwater heater strings and how reactor power will be effected as a result of the isolated feedwater heater string.

9 **10-154** **Points: 1.00**

Which of the following describes the capacity of the station batteries during a station blackout?

A. 2 hours assuming 2-EOP-LOPA-1 load shedding.

B. 2 hours during worst case loading (no 2-EOP-LOPA-1 load shed)

C. 6 hours during worst case loading (no 2-EOP-LOPA-1 load shed)

D. 6 hours assuming 2-EOP-LOPA-1 load shedding.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the battery capacity of 2 hours is when load shedding is performed per 2-EOP-LOPA-1. Incorrect in that load shedding will extend the battery to 4 hours.

B. Correct. The 125 VDC vital batteries are designed to have adequate capacity to supply its Station Blackout (SBO) coping and restoration loads during the four (4) hour SBO coping duration with manual stripping of some non-essential loads (ref. section 7.2.2.3 in VTD 314204). The non-essential load sheds allow for extending vital battery availability. Salem FSAR Section 8.1.4.1 states that the batteries are designed for LOOP only, Loss of All AC Power Is not a Design Bases Accident (DBA). NOTE: the batteries are rated for two (2) hours of operation following a Loss of All AC Power without load shed.

C. Incorrect. Plausible if the operator may incorrectly recall and confuse the 4 hour extended battery capacity with the 6 hours.

D. Incorrect. Plausible if the operator may incorrectly recall and confuse the 4 hour extended battery capacity with the 6 hours. Incorrect in that load shedding will extend the battery to 4 hours.

Technical References:	S2.OP-AB.LOOP-0001 (R32), PSBP 314204 (SBO calculation), FSAR Sect 8.1.4.1
Proposed References to be provided:	None
Learning Objective:	NOS05DCELEC-11 Objective 2. Describe the design bases of the DC Electrical System.
Question Source:	Bank – Beaver Valley 2017 NRC RO11
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	10-Station Blackout
K/A #:	EPE 55 K3.01 Knowledge of the reasons for the following responses and/or actions as they apply to Station Blackout: Length of time for which battery capacity is designed.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant must have knowledge of the design capacity of the 125 VDC vital batteries during an SBO event.

10 **11-204** **Points: 1.00**

Given:

- Salem Unit 2 experienced a Loss of Off-Site Power
- The crew is performing S2.OP-AB.LOOP-0001, Loss of All Offsite Power
- ALL Charging Pumps on Unit 2 are unavailable
- 13 Charging Pump from Unit 1 is being aligned to supply charging flow to Unit 2

Then:

- Power to the 4KV Group Busses have been restored
- Preparations for re-starting RCPs are in progress
- RCP seal inlet temperatures are;
    - 21 RCP - 227 °F
    - 22 RCP - 220 °F
    - 23 RCP - 219 °F
    - 24 RCP - 221 °F

In accordance with S2.OP-AB.LOOP-0001, which of the following completes the statement concerning restoring RCP seal injection flow?

Based on the above condition, restoring seal injection flow to the RCPs __(1)__ permitted. The reason for checking RCP seal inlet temperatures is to prevent damage to the RCP ___(2)___.

A. (1) is NOT (2) pump bearings

B. (1) is (2) pump bearings

C. (1) is (2) seals

D. (1) is NOT (2) seals

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the RCP pump bearings is the component of concern.

B. Incorrect. Plausible because the operator may believe that the RCP pump bearings is the component of concern.

C. Incorrect. Plausible because the operator may misinterpret the note in S2.OP-AB.LOOP-0001 Attachment 10 and believe that seal injection can be restored unless ALL RCP seal inlet temperatures are ≥ 225 °F.

D. Correct. Per S2.OP-AB.LOOP-0001, Attachment 10, if ANY one of the RCP seal inlet temperatures is ≥ 225 °F, then restoring seal injection flow to the RCPs is NOT permitted. Based on 21 RCP seal inlet temperature is ≥ 225 °F, then restoring seal injection flow to the RCPs is NOT permitted and in order to supply charging from 13 charging pump, Unit 2 must isolate seal injection to the RCPs prior to starting 13 charging pump. The reason for this is to prevent thermal shock from restoring seal injection and damage to the RCP seals.

Technical References:	S2.OP-AB.LOOP-0001 (R34)
Proposed References to be provided:	None
Learning Objective:	NOS05ABLOP1-09, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.LOOP-0001 and the bases for the actions
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-Loss of Offsite Power
K/A #:	APE 56 AK3.02 Knowledge of the reasons for the following response and/or actions as they apply to Loss of Offsite Power: Actions contained in AOPs
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant the reason why seal injection is not restored based on seal inlet temperatures.

Additional Reference:

S2.OP-AB.LOOP-0001(Q)

ATTACHMENT 10
(Page 1 of 2)

PLACING 13 CHARGING PUMP INSERVICE
TO UNIT 2 FROM THE UNIT 1 RWST

NOTE

___ When utilizing 13 Charging Pump, Unit 1 should monitor RWST level to ensure compliance with Technical Specification 3.5.5. If Unit 1 RWST approaches the minimum volume required to satisfy Technical Specification 3.5.5, then either `continued operation of the 13 Charging Pump should be evaluated or appropriate compensatory actions implemented.

___ 1.0 IF **ANY** 21-24 RCP Seal Inlet Temperature is greater than or equal to 225°F, THEN **PERFORM** the following to establish charging flow ONLY to the RCS for inventory control:

___ ♦ **CLOSE** 2CV83, SEAL WATER FILTER INLET.

___ ♦ **CLOSE** 2CV89, SEAL WATER FILTER INLET.

___ ♦ **CLOSE** 2CV95, SEAL WATER FILTER BYPASS.

___ 2.0 **ENSURE** 13 Charging Pump is aligned for Appendix R Only (Unit 2) IAW S1.OP-SO.CVC-0002(Q), Charging Pump Operation.

Attachment 10 provides instructions for aligning 13 Charging Pump to Unit 1 when all Unit 2 Charging Pumps are unavailable. This attachment aligns 13 Charging Pump via the CVCS Cross-Connect System from the Unit 1 RWST to supply charging flow for RCS inventory control and boration capability to facilitate cooldown to Mode 5, and RCP seal injection after all RCP seal inlet temperatures are verified to be less than 225°F. If ANY 21-24 RCP Seal Inlet Temperature is greater than or equal to 225°F, then 13 Charging Pump is NOT started until ALL Unit 2 RCP seal injection isolation valves are closed to prevent thermal shock of the RCP seals when 13 Charging Pump is started.

11 **11-205** **Points: 1.00**

Given:

- Salem Unit 1 is at 8% power performing a reactor startup

Then:

- 1B Vital Instrument Bus de-energizes

Which of the following describes the resulting plant response?

A. The reactor does NOT trip because the Nuclear Instrumentation monitors will fail low.

B. The reactor does NOT trip because power is still below P-10.

C. A reactor trip occurs because one Intermediate Range channel bi-stable de-energizes.

D. A reactor trip occurs because one Source Range channel bi-stable de-energizes.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the NI channels will fail low on loss of power, however, on loss of power the affected channel bi-stables will trip sending a signal to SSPS trip logic.

B. Incorrect. Plausible because the operator may believe that with reactor power above P-6 the SR and IT channels are blocked. Incorrect in that the IR high flux is blocked at P-10 (10%).

C. Correct. When the 1B vital instrument bus de-energizes the 2N32 (SR), 2N36 (IR) and 2N41 (PR) channels will lose instrument and control power and the monitors will fail low. In addition, on the loss of instrument power the affect NI channels will cause the bi-stables to SSPS to trip resulting in a trip signal to the SSPS trip logic. In this case with reactor power above P-6 (1E-5%) but below P-10 (10%), the reactor will trip on 1/2 IR channel high flux bi-stables tripped. The SR channels are blocked above P-6 and the PR reactor trip is based on 2/4 PR channels high flux. The IR high flux channels are block above P-10.

D. Incorrect. Plausible because the operator may incorrectly recall when the SR channels are blocked and believe that the loss of power to 2N32 will cause a reactor trip based on 1/2 SR channel bi-stables tripped. Incorrect in that the SR channels are blocked above P-6.

Technical References:	221052, 221053
Proposed References to be provided:	None
Learning Objective:	NOS05EXCORE-14, Objective 12.b. Given plant conditions, relate the Excore Nuclear Instrumentation System with the following: 115VAC Electrical System
Question Source:	Modified Bank - Byron Oct 2019 NRC RO58
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	11-Loss of Vital AC Instrument Bus
K/A #:	APE 57 AK2.08 Knowledge of the relationship between Loss of Vital AC Instrument Bus and the following systems or components: NI.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator on how the loss of vital instrument power will affect the Nis. In this case the operator is being asked whether a reactor trip will occur when instrument power is lost requiring knowledge of reactor trip setpoints, coincidence, and permissives.

Parent Question:

Modified question based on the answer being different for Salem's design.

58 **ID: 2019 NRC-2 Q 58** **Points: 1.00**

Unit 1 is at 8% power during a plant startup.

Instrument bus 111 BYPASS SOURCE TO LOAD pushbutton is pressed.

What is the resulting plant response?

A. The reactor does NOT trip because power is rapidly transferred to the CVT.

B. The reactor does NOT trip because power is still below P-10.

C. A reactor trip occurs because N35 momentarily de-energizes.

D. A reactor trip occurs because N31 momentarily de-energizes.

Answer: A

12 **11-206** **Points: 1.00**

Given

- Salem Unit 2 experienced a loss of the 2A 125 VDC bus

Then:

- A Safety Injection actuation occurs

Based on the above conditions, which of the following completes the statement describing the remote operation of the 2A EDG and why?

The 2A EDG...

A. will NOT start because the EDG start circuit is supplied from DC power.

B. will NOT start because the SEC will not have power to energize the diesel start relay.

C. will start because SSPS directly starts the EDG on an SI and has redundant DC power supplies.

D. will start because the EDG start circuit is supplied from vital instrument power.

Answer: A

Explanation / Justification:

A. Correct. The EDG starting circuits are supplied from 125 VDC power. With no DC power, the EDG will not be able to start automatically from an SI/SEC signal or remotely from the control room.

B. Incorrect. Plausible because the operator may incorrectly recall that the SEC start relays are supplied from DC power. Incorrect in that the SEC relays are supplied by vital instrument power.

C. Incorrect. Plausible because SSPS does send a signal that would start the EDGs. Incorrect in that SSPS sends an SI signal to the SECs which the SECs will send a start signal to the EDG start circuitry which is DC powered.

D. Incorrect. Plausible because vital instrument power is used in many control circuits. Incorrect in that the EDG start circuit requires DC power.

Technical References:	NOS05EDG000-15
Proposed References to be provided:	None
Learning Objective:	NOS05EDG000-15, Objective 14.c. Given plant conditions, relate the Emergency Diesel Generator with the following: DC Distribution System.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-Loss of DC Power
K/A #:	APE 58 AK3.01 Knowledge of the reasons for the following responses and/or actions as they apply to Loss of DC Power: Operation of the EDGs.
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the questions is asking the operator how the loss of DC power will affect the starting of the EDG and the reason why.

13 **11-207** **Points: 1.00**

Which of the following completes both statements concerning the design feature of the 2R13 radiation monitors?

1. The 2R13 radiation monitors are designed to monitor the radioactive effluent release pathway from the __(1)__.
2. A loss of the __(2)__ Circulating Water system flow will prevent monitoring of the effluent pathway.

A. (1) CFCUs
(2) Unit 1

B. (1) CFCUs
(2) Unit 2

C. (1) CCW Heat Exchangers
(2) Unit 2

D. (1) CCW Heat Exchangers
(2) Unit 1

Answer: A

Explanation / Justification:

A. Correct. The discharge of the CFCUs are monitored by the R13s for radioactivity from containment into the SW system. The CFCUs SW flow discharges into the opposite units Circulating Water system.

B. Incorrect. Plausible because the operator would expected the discharge of the CFCUs to go to its respective CW system.

C. Incorrect. Plausible because SW does flow through the CCW HXs and could believe that an effluent pathway may exist from the RHR HXs to SW system. Incorrect that this is not the design purpose of the R13s.

D. Incorrect. Plausible because SW does flow through the CCW HXs and could believe that an effluent pathway may exist from the RHR HXs to SW system. Incorrect that this is not the design purpose of the R13s.

Technical References:	S2.OP-SO.RM-0001, ODCM 3.3.8
Proposed References to be provided:	None
Learning Objective:	NOS05RMS000-20, Objective 3.i. Describe how the following components impact the Radiation Monitoring System during normal and abnormal conditions: R13A and B, CFCU Service Water Monitors
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	11- Loss of Nuclear Service Water
K/A #:	APE 62 AK2.08 Knowledge of the relationship between Loss of Service Water and the following system or components: PRMS
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the loss of service water will affect the operability of the 2R13s radiation process monitors.

Additional References: S2.OP-SO.RM-0001 and ODCM 3.3.8

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that:

a. At least once per 8 hours, local monitor readouts for the affected channels are verified to be below their alarm setpoints, or
b. With a Service Water System leak (inside containment) on the Containment Fan Coil Unit associated with the inoperable monitor either:
   1. At least once per 8 hours, grab samples are to be collected and analyzed for principal gamma emitters, I-131, and dissolved and entrained gases at the lower limits of detection specified in ODCM CONTROL Table 4.11-1.B, and the ODCM Surveillance Requirement 4.11.1.1.2 is performed, or
   2. Isolate the release pathway.
c. With no identified service water leakage (inside containment) on the Containment Fan Coil Unit associated with the inoperable monitor, at least once per 24 hours, collect grab samples and analyze for principal gamma emitters, I-131, and dissolved and entrained gases at the lower limits of detection specified in ODCM CONTROL Table 4.11-1.B, and the ODCM Surveillance Requirement 4.11.1.1.2 is performed.

14 **11-208** **Points: 1.00**

Given:

- Salem Unit 2 experienced reactor trip and Safety Injection due to an Inadvertent SI
- All systems were in normal alignment and in AUTO prior to the reactor trip
- Instrument air to containment is isolated
- PZR PORVs are cycling on the air accumulators to maintain PZR pressure

Then:

- Instrument Air to containment is restored

Which of the following describes the status of the PZR PORVs and spray valves when air to containment is restored?

PZR PORVs air accumulators ___(1)___ automatically align back its normal air supply header and the PZR spray valves ___(2)___ automatically re-open when air is restored.

A. (1) will
(2) will NOT

B. (1) will NOT
(2) will NOT

C. (1) will NOT
(2) will

D. (1) will
(2) will

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that on loss of air in containment that the PZR spray valves will have to be manually re-opened.

B. Incorrect. Plausible because the operator may believe that operator action is required in containment to realign the accumulators back to the normal air supply.

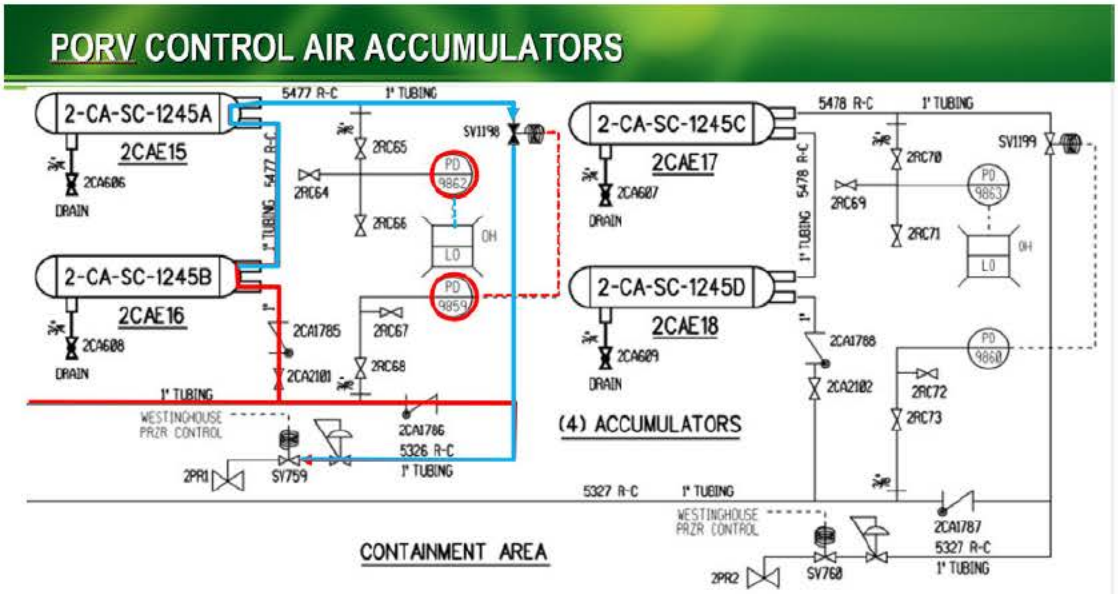
C. Incorrect. Plausible because the operator may believe that operator action is required in containment to realign the accumulators back to the normal air supply.

D. Correct. The PZR PORV air accumulators will automatically isolate and be supplied by the normal containment control air when air pressure in containment is restored and the Master Pressure Controller will have a demand to open spray valves, therefore when air to containment is restored both spray valves will re-open.

Technical References:	NOS05CONAIR-13
Proposed References to be provided:	None
Learning Objective:	NOS05CONAIR-13, Objective 4.b. Describe how each impact the Control Air System during normal and abnormal conditions: PORV Control Air Accumulators
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	11- Loss of Instrument Air
K/A #:	APE 65 AA1.03 Ability to operate and/or monitor the following as they apply to Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained
Importance:	3.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the PZR PORV control air accumulators realign following restoring control air to containment and whether the spray valves will re-open.

Additional Reference:



Normal Air is available - > 90 psig

Accumulator aligned - < 85 psig

15 **11-209** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power with 350 MVARs OUT (lagging)
- Salem Unit 2 is at 100% power with 200 MVARs OUT (lagging)

Then:

- Salem Unit 1 trips

Based on the above condition, which of the following completes the statement describing the status of the 500 KV grid?

Grid voltage will __(1)__ and Salem Unit 2 MVARs will __(2)__.

A. (1) rise
(2) rise

B. (1) rise
(2) lower

C. (1) lower
(2) lower

D. (1) lower
(2) rise

Answer: D

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that a unit trip will rise grid voltage. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that a unit trip will rise grid voltage. Second part is incorrect. Plausible if the operator confuses the concept on how grid MVARs respond to grid voltage and believes that a lower grid voltage will cause a lowering of MVARs.

C. Incorrect. First part is correct. Second part is incorrect. Plausible if the operator confuses the concept on how grid MVARs respond to grid voltage and believes that a lower grid voltage will cause a lowering of MVARs.

D. Correct. Per S2.OP-AB.GRID-0001, Note for step 1.4, a loss of either Salem Unit 1 or 2 may result in the 500KV switchyard voltage lowering below 493 KV or by 2%. As the grid voltage lowers, the generating units on the grid will experience sharing of the reactive load and result in, in this case, Salem Unit 1 MVARs rising following the trip of Salem Unit 2.

Technical References:	S2.OP-AB.GRID-0001 (R23)
Proposed References to be provided:	None
Learning Objective:	NOS05ABGRID-13, Objectives 3.b. Given a set of initial plant conditions: Describe the plant response to actions taken in the abnormal procedure.
Question Source:	Modified – Salem 2022 NRC RO17
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	11-Generator Voltage and Electric Grid Disturbances
K/A #:	APE 77 AA2.05 Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Status of grid
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator the effect on the grid following a trip or loss of a Salem unit. In addition, the operator is being asked how the MVARs (reactive load) will be affected following the loss of Salem Unit 1.

Parent Question: Salem 2022 NRC RO17

Modified Part 2 of the question to test what happens to grid voltage.

Given:

- Both Units 1 and 2 are at 100% power with Voltage Regulators in AUTO.
- Unit 1 reactive load is 250 MVARs OUT.
- Unit 2 reactive load is 250 MVARs OUT.
- The 500 KV Ring Bus is closed.

Subsequently, Unit 2 trips.

Which ONE of the following completes both statements?

1. Unit 1 Main Generator Reactive (MVAR) loading will __(1)__.

2. Generator field current will __(2)__.

A. 1) rise by less than 250 MVAR
   2) rise

B. 1) rise by less than 250 MVAR
   2) lower

C. 1) rise 250 MVAR
   2) rise

D. 1) rise 250 MVAR
   2) lower

16 **10-029** **Points: 1.00**

Given:

- Salem Unit 2 is responding to a Loss of Coolant Accident per 2-EOP-LOCA-6, LOCA Outside Containment.
- ALL ECCS Pumps are running
- The crew suspects a leak on an SI Pump discharge Cold Leg
- RCS pressure is 1700 psig and steady

Which of the following indications would corroborate the leak and the valve(s) required to be closed to isolate the leak per 2-EOP-LOCA-6?

A. SI Pump flows indicating 200 gpm; 2SJ135, Cold Leg Discharge Valve

B. SI Pump flows indicating 200 gpm; 21 and 22 SJ49, Cold Leg Isolation Valves

C. SI Pump pressure at 1700 psig; 21 and 22 SJ49, Cold Leg Isolation Valves

D. SI Pump pressure at 1700 psig; 2SJ135, Cold Leg Discharge Valve

Answer: A

Explanation / Justification:

A. Correct. An indicate SI Pump flow 200 gpm with RCS pressure at 1700 psig would indicate a leak on the SI Cold Leg. RCS pressure at 1700 psig is above the shutoff head of an SI Pump. Therefore, flow indicated on the SI Pumps indicates that a leak exists on the cold leg(s). Per 2-EOP-LOCA-6 step 2, the 2SJ135 Cold Leg Discharge Valve is closed to isolate a leak on either of the SI Pump Cold Legs.

B. Incorrect. Plausible because 21 and 22 SJ49 are clod leg isolation valves and are directed to be closed in EOP-LOCA-6. Incorrect in that the SJ49s are used to isolate a leak on the RHR cold legs.

C. Incorrect. Plausible because 21 and 22 SJ49 are clod leg isolation valves and are directed to be closed in EOP-LOCA-6. Incorrect in that the SJ49s are used to isolate a leak on the RHR cold legs. Also the operator may believe that when the SI discharge pressure reads about the same RCS pressure that a leak exists.

D. Incorrect. Plausible because the 2SJ135 is closed per EOP-LOCA-6. Incorrect in that SI Pump discharge pressure alone does not necessarily indicate a leak, SI Pump flow is the best indicator because the RCS pressure is above the shutoff head of the SI pumps.

Technical References:	2-EOP-LOCA-6 (R41) & bases document
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA06-05, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in the EOP for LOCA OUTSIDE CONTAINMENT
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.5 to 45.8
Level:	RO
Category/System:	10-LOCA Outside Containment
K/A #:	W E04 EA1.04 Ability to operate and/or monitor the following as they apply to LOCA Outside Containment: RCS
Importance:	3.8
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator how to determine a leak on the SI Pump Cold Leg based on RCS pressure and what valve is closed per EOP-LOCA-6 to isolate the leak on the SI Pump Cold Leg location.

17 **10-155** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA

Then:

- Loss of Emergency Recirculation exist
- The crew is implementing 2-EOP-LOCA-5, Loss of Emergency Recirculation
- RCS depressurization is in progress

Per 2-EOP-LOCA-5, which of the following completes both statements?

3. During the RCS depressurization upper head voiding __(1)__ expected.

4. Which of the following indications is monitored if head voiding occurs?

A. (1) is
(2) Core Exit Thermocouples

B. (1) is
(2) PZR level

C. (1) is NOT
(2) PZR level

D. (1) is NOT
(2) Core Exit Thermocouples

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because CET temperatures may increase due to head voiding. Incorrect in that EOP-LOCA-5 step 23 Note states that rapidly rising PZR level is a result of upper head voiding.

B. Correct. Per 2-EOP-LOCA-5 step 23, during RCS depressurization upper head voiding can be expected. As such, the operators need to monitor a rise in PZR level in order to stop the depressurization prior to the PZR going solid.

C. Incorrect. Plausible because the operator may believe that as long as the cooldown rate of 100 °F per hour is not exceeded then no possibility of upper head voiding will be expected.

D. Incorrect. Plausible because the operator may believe that as long as the cooldown rate of 100 °F per hour is not exceeded then no possibility of upper head voiding will be expected. Plausible because CET temperatures may increase due to head voiding. Incorrect in that EOP-LOCA-5 step 23 Note states that rapidly rising PZR level is a result of upper head voiding.

Technical References:	2-EOP-LOCA-5 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA05-06, Objective 6. Given any step, caution, note, or Continuous Action Summary for LOSS OF EMERGENCY COOLANT RECIRCULATION, describe the basis.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10- Loss of Emergency Coolant Recirculation
K/A #:	W EPE11 EK1.07 Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Emergency Coolant Recirculation: Upper head voiding during RCS depressurization
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the applicant is being asked if upper head voiding is expected during the RCS depressurization and what parameter will indicate the presence of voiding.

Additional Reference:

EOP Step No: Note 23-1

ERG Step No: Note 23-1

EOP Step:

NOTE: THE UPPER HEAD REGION MAY VOID DURING RCS DEPRESSURIZATION IF RCPS ARE NOT RUNNING. THIS WILL RESULT IN A RAPIDLY RISING PZR LEVEL
[RCS DEPRESSURIZATION TO MINIMIZE SUBCOOLING]

Purpose:

To inform the operator that PZR level may increase rapidly during RCS depressurization with no RCPs running due to voiding in the upper head region.

ERG Basis:

Without RCPs running, there is very little flow into the upper head region. Liquid in that region remains relatively hot even though the liquid temperature in the active regions of the RCS has been significantly reduced during the RCS cooldown. As the RCS is subsequently depressurized, the hotter liquid in the upper head may flash to steam, forming a upper head void. Steam formation in the upper head will displace water into the PZR, causing rapidly increasing PZR level with the potential for water relief through the PZR PORVs. The PZR may fill with water within a few minutes. This note informs the operator of the potential for this condition, so that RCS depressurization can be stopped quickly to avoid a water solid PZR.

18 **10-156** **Points: 1.00**

Given:

- Salem Unit 2 has lost ALL feedwater
- The crew is performing 2-EOP-FRHS-1, Response to Loss of Secondary Heat Sink
- Safety Injection (SI) has been manually actuated and reset
- 24 SG is selected for depressurization
- The RO actuates MSLI on all loops except for 24
- Unknown to the crew, 22MS167 failed to close
- The PO adjusts the steam pressure valve demand to commence the depressurization

Based on the above conditions and in accordance with 2-EOP-FRHS-1, which of the following is the expected response and any required action if the depressurization rate is set too HIGH?

A. No effect on the depressurization since the Main Steam Line Isolation (MSLI) signals are blocked when SI was manually actuated and reset.

B. A High Steam Line Differential Pressure will result in a MSLI signal and isolate steam dumps; the crew will use atmospheric dumps to continue the depressurization.

C. A High Steam Flow with Low-Low-Tavg will result in a MSLI signal and isolate steam dumps; the crew will use atmospheric dumps to continue the depressurization.

D. The Low-Low-Tavg (P-12) signal will close all steam dumps; the crew will depress the bypass Tavg pushbuttons to continue the depressurization using ALL steam dump groups.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly believe that manually actuating and resetting SI will block ALL MSLI signals (SI and MSLI). Incorrect in that the MSLI signal for high steam flow with low-low Tavg and hi-hi containment pressure can NOT be blocked..

B. Incorrect. Plausible because the high steamline differential pressure SI can NOT be blocked. Incorrect in that this signal does not actuate a MSLI.

C. Correct. A 100% steam dump valve dump will cause high steam flows and low-low Tavg temperature (P-12). P-12 will close the steam dumps, however, the high steam flow rate with P-12 will also actuate a MSLI closing all MSIVs resulting in no available steam dumps. The crew can continue the depressurization via the MS10s (atmospheric dumps).

D. Incorrect. Plausible because the operator may incorrectly believe that when SI was manually actuated and then reset, that MSLI signals are blocked. Incorrect in that MSLI signals can NOT be blocked.

Technical References:	2-EOP-FRHS-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRHS00-05, Objective 6.a Describe the plant response to actions taken in the following EOP's. 2-EOP-FRHS-1
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7. / 45.7 / 45.8
Level:	RO
Category/System:	10-Loss of Secondary Heat Sink
K/A #:	W E05 EK1.09 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to Loss of Secondary Heat Sink: MSLI on high steam pressure rate during cooldown.
Importance:	3.6
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the question is asking the operator knowledge of MSLI signals and the cause and effect when operators are adjusting steam dumps to establish maximum rate of depressurization which can result in an unwanted MSLI and stopping the depressurization.

19 **11-210** **Points: 1.00**

Given:

- Salem Unit 2 has experienced a reactor trip
- The crew has transitioned to 2-EOP-TRIP-2, Reactor Trip Response
- RO reports three (3) control rods are NOT FULLY inserted
- The crew initiates a Rapid Boration via a BAT Pump
- 2CV175, Rapid Boration Valve, is fully OPEN
- Rapid Boration flowmeter indicates 20 gpm

Which of the following describes the NEXT required action per 2-EOP-TRIP-2?

A. Continue with rapid boration until the required amount of boron is achieved.

B. Stop the BAT pump, realign charging pump suction to the RWST, and maintain charging flow > 87 gpm.

C. Establish rapid boration via cold leg injection by realigning charging pump suction to the RWST and opening the BIT valves.

D. Direct equipment operator to locally open the 2CV174 to bypass the boric acid blender, fully open 2CV172 for maximum boron flow, and maintain charging flow > 87 gpm.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that adequate rapid boration flow is met. Incorrect in that rapid boron flow should be 36 gpm.

B. Correct. Per 2-EOP-TRIP-2 step 4, rapid boration is initially started using a BAT pump and then opening the 2CV175. Adequate rapid boration flow through the 2CV175 is 36 gpm. Based on the indicated boron flow, rapid boration flow is inadequate. As a result EOP-TRIP-2 will stop the BAT pump, realign the charging suction from the RWST.

C. Incorrect. Plausible because this action is performed in 2-EOP-FRSM-1 if inadequate rapid boration flow. Incorrect in that EOP-TRIP-2 does not direct this pathway.

D. Incorrect. Plausible because this action is performed for rapid boration when emergency borating from the field during control room evacuation.

Technical References:	2-EOP-TRIP-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP002-08, Objective 5. Determine the indications that are monitored to ensure proper system/component operation for each step in EOP-TRIP-2
Question Source:	Bank – Byron 2019 NRC RO21
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	RO
Category/System:	11-Emergency Boration
K/A #:	APE 24 AA2.12 Ability to determine and/or interpret the following as they apply to Emergency Boration: Emergency boron flow
Importance:	3.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking to determine rapid boration flow is adequate and, if not, what is the next required action IAW EOP-TRIP-2.

Additional Reference: S2.OP-SO.CVC-0008

4.4 Boration from RWST

_____4.4.1. **DETERMINE** required Boration time (as applicable):

Condition	RWST (gals)	Time (min)
Each Stuck Out Control Rod	8395	120
10 Steps of Control Rod Bank Mis-alignment	3358	50
10 ppm RCS Boron Concentration Increase	1064	20
1°F RCS Cooldown	743	15

3 rods x 8395 per stuck rod = 25,185 gallons

RWST level 44 feet = 395,000 gallons

395,000 gals – 25,185 gals = 369,815 gals

369,818 gals approx. 41 feet in RWST

20 **10-157** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- ALL Circulators are in service
- The BOP operator notices condenser backpressure is rising, but is unsure if it's valid

What other indication in the control room would BEST confirm the rising backpressure and what is the operational concern if the conditions above continue?

A. Rising vibrations on the LP turbine; damage to the Low Pressure turbine.

B. Rising vibrations on the LP turbine; cavitation of the condensate pumps.

C. Lowering generator megawatts; cavitation of the condensate pumps.

D. Lowering generator megawatts; damage to the Low Pressure turbine.

Answer: D

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because a high backpressure may cause elevated turbine vibrations. Incorrect in that the best indication to look at is lowering MWs. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because a high backpressure may cause elevated turbine vibrations. Incorrect in that the best indication to look at is lowering MWs. Second part is incorrect. Plausible because with a higher backpressure the turbine exhaust temperatures would rise. Incorrect in that overpressurization and damage to the condenser is the primary concern.

C. Incorrect. First part is correct. Second part is incorrect. Plausible because with a higher backpressure the turbine exhaust temperatures would rise. Incorrect in that overpressurization and damage to the condenser is the primary concern.

D. Correct. Lowering condenser backpressure will cause the LP turbines to be less efficient and therefore result in a reduction or lowering of the main generator MWe output. The operational concern with a high backpressure is overpressurizing and damage to the main condenser.

Technical References:	NOS05MNTURB-14
Proposed References to be provided:	None
Learning Objective:	NOS05MNTURB-14, Objective 13.c. Given plant conditions, relate the Main Turbine System with the following: Main Condenser
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	11-Loss of Condenser Vacuum
K/A #:	APE 51 AK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to Loss of Condenser Vacuum: Relationship of condenser and M/TG operation
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked how condenser backpressure effects the output of the main turbine/generator and in addition what is the operational concern with high backpressure.

21 **10-158** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Containment Pressure Channel III is inoperable and out of service for troubleshooting, The Channel has been properly removed from service IAW S2.OP-SO.RPS-0005, Placing Containment Pressure Channel in Tripped Condition

Then:

- Salem Unit 2 reactor tripped and SI actuated due to a LOCA
- The crew is implementing 2-EOP-TRIP-1, Reactor Trip or Safety Injection

The following conditions are now present:

- Containment pressure readings are:
  - Channel I – 15.2 psig
  - Channel II – 14.7 psig
  - Channel III – 0.0 psig
  - Channel IV – 14.5 psig

Based on the above conditions, what is the status of the Containment Spray system?

1. Conditions for Containment Spray and Phase 'B' Actuation __(1)__ met.
2. The Containment Spray Pump discharge valves (CS2s) will open automatically on a __(2)__ signal.

A. (1) are NOT
(2) Containment Spray

B. (1) are NOT
(2) Phase B

C. (1) are
(2) Phase B

D. (1) are
(2) Containment Spray

Answer: A

Explanation / Justification:

A. Correct. Conditions for Containment Spray and Phase B are not met based on 1/3 containment pressure channels > 15 psig. In this case with Channel III removed from service Channel III is bypassed which makes the logic 2/3 containment pressure channels are above 15 psig. Only Channel I is above the 15 psig therefore, no CS signal. The CS pump discharge valves (CS2s) open automatically on a CS actuation signal only.

B. Incorrect. Plausible because Phase B signal is actuated at the same 2/4 containment pressure logic. Incorrect in that only the CS signal actuates the CS valves.

C. Incorrect. Plausible because the operator may incorrectly recall the logic coincidence for CS spray and believe that you need 1/3 containment pressure signals present to actuate CS and Phase B based on Channel III removed from service. Plausible because Phase B signal is actuated at the same 2/4 containment pressure logic. Incorrect in that only the CS signal actuates the CS valves and when a containment pressure channel is removed from service, the channel is bypassed so that the logic goes from 2/4 to 2/3.

D. Incorrect. Plausible because the operator may incorrectly recall the logic coincidence for CS spray and believe that you need 1/3 containment pressure signals present to actuate CS and Phase B.

Technical References:	2-EOP-TRIP-1 (R42), NOS05CSPRAY-07
Proposed References to be provided:	None
Learning Objective:	NOS05CSPRAY-07, Objective 15.a. Given plant conditions relate the Containment Spray System with the following: RPS
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.7 / 45.7 / 45.8
Level:	RO
Category/System:	10-High Containment Pressure
K/A #:	W E14 EA1.08 Ability to operate and/or monitor the following as they apply to High Containment Pressure: CCS
Importance:	4.0
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the applicant to monitor the containment pressure channels and determine if CS and Phase B have actuated and then determine which of the signals; CS or Phase B will open the CS pump discharge valves (CS2s).

22 **11-211** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Indications of High RCS Activity exists
- The crew is implementing S2.OP-AB.RC-0002, High Activity in the RCS
- Chemistry sample of the RCS confirms the elevated RCS activity
- The RCS activity is <u>below</u> the TS limit

Which of the following describes the NEXT required action and the reason for the action?

A. Reduce letdown flow to minimize personnel radiation exposure in the Auxiliary Building.

B. Perform hourly RCS sampling to determine if TS limits are exceeded and a shutdown is required.

C. Maximize letdown flow to accelerate RCS activity cleanup through the demineralizers.

D. Shutdown the Unit within 6 hours to minimize the potential to release high radioactive airborne during a SG tube rupture.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because reducing letdown flow could reduce the dose rates in the Aux Building. Incorrect in that S2.OP-AB.RC-0002 does not direct reducing letdown flow, just the opposite letdown flow is increased.

B. Incorrect. Plausible because step 3.20 of S2.OP-AB.RC-0002 directs hourly sampling. Incorrect in that maximum letdown would be the next action by procedure..

C. Correct. Per S2.OP-AB.RC-0002 following confirmation of the RCS elevated activity and being below TS limits, the crew will maximize letdown flow to accelerate RCS cleanup through the demineralizers.

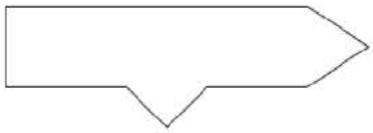
D. Incorrect. Plausible because this is the bases for TS 3.4.9, Specific Activity, to limit release during SGTR with primary-to-secondary leakage.

Technical References:	S2.OP-AB.RC-0002 (R12)
Proposed References to be provided:	None
Learning Objective:	NOS05ABRC02-07, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.RC-0002 and the bases for the actions in accordance with the Technical Bases Document.
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	11-High Reactor Coolant Activity
K/A #:	APE 76 AK3.04 Knowledge of the reason for the following response and/or actions as they apply to High Reactor Coolant Activity: Maximizing demineralizer flow rates
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked the reason for maximizing letdown flow rates through the demineralizers during elevated RCS activity.

23 **10-160** **Points: 1.00**

When proceeding through 2-EOP-TRIP-3, SI Termination, you come across the following symbol at Step 1.



What does the symbol denote?

A. The stated action must be completed before continuing with the present flow path.

B. Initiate the stated action while continuing with the existing flow path.

C. The stated action shall be executed if or when the conditions specified occurs.

D. The crew must wait for the conditions specified to become established before proceeding on.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because this describes a "Sequential Branching" symbol which are used in EOPs.

B. Correct. Per OP-SA-108-101-2000 Figure 2, Flowchart Symbols, this symbol indicates a Concurrent Step symbol which requires the operator to enter the designated procedure and perform the stated actions while continuing in the existing flow path.

C. Incorrect. Plausible because this describes a "Conditional Step" symbol which are used in EOPs.

D. Incorrect. Plausible because this describes a "Wait Step" symbol which are used in EOPs.

Technical References:	OP-SA-108-101-2000 (R12)
Proposed References to be provided:	**Embedded symbol from EOP-TRIP-3**
Learning Objective:	NOS05CONDOP-15, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.13
Level:	RO
Category/System:	10-Rediagnosis & SI Termination / Emergency Procedures/Plan
K/A #:	W E01 & E02 G2.4.29 Knowledge of emergency and abnormal operating procedures layout, symbols, and icons.
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked knowledge of symbols used on EOPs and how the crew should implement them.

24 **10-161** **Points: 1.00**

Given:

- Salem Unit 1 experienced a reactor trip and automatic actuation of SI due to a LOCA.
- Containment pressure peaked at 20 psig and is now 13 psig and lowering
- The crew is continuing with 1-EOP-LOCA-1, Loss of Reactor or Secondary Coolant, following the successful transfer to Cold Leg recirculation
- Due to containment high water level, the crew has entered 1-EOP-FRCE-2, Response to Containment Flooding

Which of the following systems is the cause of the unexpected water level in containment?

A. Fire Protection

B. Service Water

C. Component Cooling Water

D. Primary Water

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the Fire Protection system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that FP system is isolated on Phase A signal.

B. Correct. Service Water from the CFCUs does not get isolated by Phase A or B signals. Therefore, SW system is the only source of water into containment following a Phase A and B signals.

C. Incorrect. Plausible because the Component Cooling Water system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that CCW system is isolated on Phase B signal since containment pressure greater than 15 psig

D. Incorrect. Plausible because the Primary Water system is a source of water into containment per 1-EOP-FRCE-2. Incorrect in that PW system is isolated on Phase A signal.

Technical References:	1-EOP-FRCE-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCE00-08, Objective 1.2. Describe the EOP mitigation strategy for the following: 1-EOP-FRCE-2
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.8 / 41.10 / 45.3
Level:	RO
Category/System:	10-Containment Flooding
K/A #:	W E15 EK2.03 Knowledge of the relationship between Containment Flooding and the following systems or components: Service water
Importance:	3.3
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked the sources of water into containment that could be causing flooding in containment. The applicant also needs to know which system is not isolated when a Phase A and B signals are actuated.

25 **10-162** **Points: 1.00**

Given:

- Salem Unit 2 experienced a reactor trip from 100% power

Then:

- STA reports a valid CFST YELLOW path for Containment Environment based on high radiation readings in containment
- The CRS enters 2-EOP-FRCE-3, Response to High Containment Radiation Levels

Which of the following completes the statement concerning the mitigating strategy that the 2-EOP-FRCE-3 will direct?

Operate ALL CFCUs in __(1)__ speed through the __(2)__ filters to reduce the radioactivity in the containment atmosphere.

A. (1) High
(2) Roughing

B. (1) High
(2) HEPA

C. (1) Low
(2) HEPA

D. (1) Low
(2) Roughing

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the CFCUs can be operated in HIGH speed and the operator may believe that in HIGH speed higher air flow will pass thru the filters for better cleanup.

B. Incorrect. Plausible because the CFCUs can be operated in HIGH speed and the operator may incorrectly believe that the dampers to redirect the air flow thru the HEPA filters can be operated from the control room. Incorrect in that the dampers positions on the console is indication only to tell the operators if the HEPA filters is in service.

C. Correct. Per 2-EOP-FRCE-3 step 2, the crew is directed to place all CFCUs is LOW speed. In LOW speed operation the dampers will realign flow through the HEPA filters and bypass the roughing filters to reduce the radioactivity inside containment.

D. Incorrect. Plausible because LOW speed is the required operation of the CFCUs. The operator may confuse which type of filter is used. During LOW and HIGH speed operation.

Technical References:	2-EOP-FRCE03 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCE00-08, Objective 3. Identify possible radioactivity release paths for a containment high pressure or high radiation condition, and describe how the actions in 2-EOP-FRCE-1 and 3 minimize the release potential.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 41.8 / 45.5 to 45.8
Level:	RO
Category/System:	11-High Containment Radiation
K/A #:	W E16 EA1.05 Ability to operate and/or monitor the following as they apply to High Containment Radiation: Containment atmosphere filtration system
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the applicant how will the CFCUs be operated in order to place the atmosphere filtration system in service (HEPA).

26 **11-212** **Points: 1.00**

Given:

- A Loss of All Off-site power has occurred at Salem Unit 2.
- The crew is performing 2-EOP-TRIP-4, Natural Circulation.

Then:

- Power has been restored to all the 4KV Group Buses.

Per 2-EOP-TRIP-4, how many RCPs are started and what is the primary reason for starting the RCP(s) during natural circulation?

A. One; Prevents two phase flow from occurring and impeding natural circulation.

B. Two; Prevents two phase flow from occurring and impeding natural circulation.

C. Two; Permits a faster plant cooldown with less potential for upper head voiding.

D. One; Permits a faster plant cooldown with less potential for upper head voiding.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because this would be the reason for stopping an RCP during small break LOCAs. Incorrect in that this is not the primary reason stated in EOP-TRIP-4 bases document.

B. Incorrect. Plausible because two RCPs could be started for better spray if 23 RCP is not available. Incorrect in that EOP-TRIP-4 Note N1-1 does allow two RCPs to start to support better spray if 23 RCP is not available. Plausible because this would be the reason for stopping an RCP during small break LOCAs. Incorrect in that this is not the primary reason stated in EOP-TRIP-4 bases document.

C. Incorrect. Plausible because two RCPs could be started for better spray if 23 RCP is not available. Incorrect in that EOP-TRIP-4 Note N1-1 does allow two RCPs to start to support better spray if 23 RCP is not available.

D. Correct. Per 2-EOP-TRIP-4 Bases Document Note (N1-3), RCPs would be started whenever possible during the course of EOP-TRIP-4. Forced convection cooling permits a faster plant cooldown with less potential for upper head voiding, an attempt to restart RCPs should be made when under natural circulation conditions.

Technical References:	2-EOP-TRIP-4 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP004-08, Objective 4. Determine the basis for each step, caution, note or continuous action step relative to a NATURAL CIRCULATION COOLDOWN
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.10 / 45.6 / 45.13
Level:	RO
Category/System:	10-Natural Circulation
K/A #:	W E09 EK3.06 Knowledge of the reasons for the following responses and/or actions as they apply to Natural Circulation Operation: Starting an RCP
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the candidate is being asked how the reason for starting an RCP during natural circulation conditions per EOP-TRIP-4.

27 **11-213** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- OHA D-20, 21 RCP BRG CLG WTR FLO LO, is Alarming
- RO reports that all RCP motor bearing temperatures are rising

Which of the following completes the statement?

The crew has up to __(1)__ minutes to restore CCW flow OR is required to trip the reactor and stop the RCP anytime RCP motor bearing temperatures exceeds __(2)__.

A. (1) 2
(2) 175 °F

B. (1) 2
(2) 165 °F

C. (1) 5
(2) 165 °F

D. (1) 5
(2) 175 °F

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse the 2 minutes requirement with the loss of seal injection flow and loss of CCW flow to the thermal barrier.

B. Incorrect. Plausible because 165 °F is the limit in the ARP for OHA D-20 that directs transition to S2.OP-AB.RCP-0001. Also the operator may confuse the 2 minutes requirement with the loss of seal injection flow and loss of CCW flow to the thermal barrier.

C. Incorrect. Plausible because 165 °F is the limit in the ARP for OHA D-20 that directs transition to S2.OP-AB.RCP-0001.

D. Correct. Per S2.OP-AB.RCP-0001, Attachment 1, the crew has up to 5 minutes to restore the loss of CCW or a reactor trip is required and the RCP must be stopped. Also anytime if RCP motor bearing temperatures exceed 175 °F, then the reactor must be tripped and the affected RCP stopped.

Technical References:	S2.OP-AB.RCP-0001 (R28)
Proposed References to be provided:	None
Learning Objective:	NOS05RCPUMP-18, Objective 10.b, State the Technical Specification associated with the component, parameters and operation of the Reactor Coolant Pump including the Limiting Condition for Operation(s) (LCO) and the applicability of the LCO(s)
Question Source:	Bank – Beaver Valley 2018 NRC RO29
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.12 / 45.13
Level:	RO
Category/System:	11-Reactor Coolant Pump System / Conduct of Operations
K/A #:	003 G2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.
Importance:	4.4
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to evaluate performance data from the RCP seal injection surveillance test and then determine if the parameters recorded from the test meets the TS LCO.

Additional Reference:

s2.OP-AB.RCP-0001(Q)

ATTACHMENT 1
(Page 1 of 2)

CONTINUOUS ACTION SUMMARY

1.0 IF AT ANY TIME, any of the following validated RCP conditions exist, THEN GO TO Attachment 2, Stopping Reactor Coolant Pumps.

_____
Time

- Component Cooling Water flow is lost to ALL RCPs.
- Component Cooling Water flow not restored within 5 minutes of initial loss:
  - OHA-D-20, 21 RCP BRG CLG WTR FLO LO
  - OHA-D-21, 22 RCP BRG CLG WTR FLO LO
  - OHA-D-22, 23 RCP BRG CLG WTR FLO LO
  - OHA-D-23, 24 RCP BRG CLG WTR FLO LO
- RCP Seal Injection Flow AND RCP Thermal Barrier Component Cooling flows are lost concurrently (RCPs should be secured within 2 minutes to prevent RCP damage).
- Shaft vibration greater than 20 mils or motor flange vibration greater than 5 mils **[C0379]**
- Motor Bearing temperature greater than 175°F
- Motor Winding temperature greater than 302°F, (Modes 1-5)
  (Also, REFER TO Attachment 3 (page 2 of 2) for Alternate Indications to validate indicated rise in RCP Motor Winding Temperature)
- Seal Water Outlet temperature greater than 190°F
- #1 Seal Water Leakoff flow less than 0.8 gpm and reverses to greater than or equal to 6.0 gpm or pump bearing/seal inlet temperature rise
- #1 Seal Water Leakoff flow less than 0.8 gpm with rising pump bearing/seal inlet temperatures
- #1 Seal Water Leakoff flow greater than or equal to 6 gpm

28 **01-017** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 2CC71, Letdown HX Temp Control Valve, is in Auto

Then:

- 2TE-130,, temperature detector for 2CC71, fails LOW

Which of the following describes the impact of this failure?

1. Letdown temperature will __(1)__ and the RCS T-avg temperature will __(2)__.

A. (1) rise (2) lower

B. (1) rise (2) rise

C. (1) lower (2) rise

D. (1) lower (2) lower

Answer: A

Explanation / Justification:

A. Correct. When the 2TE-130 temp detector to the 2CC71 fails low, it will cause the 2CC71 to go closed and the Letdown HX outlet temperature will rise, the hotter letdown flowing through the Mixed Bed Demineralizers will cause boron to be released into the RCS resulting in a boration effect and lowering T-avg temperature..

B. Incorrect. Plausible because the operator may incorrectly recall how temperature affects the boron affinity of a mixed bed demineralizer and believe a higher temp absorbs boron.

C. Incorrect. Plausible because the operator may confuse the operation of the 2CC71 when the temperature detector fails low and believe that the 2CC71 will open. Plausible because the operator may incorrectly recall how temperature affects the boron affinity of a mixed bed demineralizer and believe a higher temp absorbs boron

D. Incorrect. Plausible because the operator may confuse the operation of the 2CC71 when the temperature detector fails low and believe that the 2CC71 will open.

Technical References:	INPO 191007 Demineralizers
Proposed References to be provided:	None
Learning Objective:	NOS05DEMINE-02, INPO 191007
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 45.5
Level:	RO
Category/System:	01-CVCS
K/A #:	004 A3.06 Ability to monitor automatic features of the CVCS system, including: T-ave and T-ref
Importance:	3.8
Tier/Group	Plant Systems - Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the Auto function of the 2CC71 is affected when the temp detector fails low and how T-ave is affected.

29 **02-031** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21 Charging Pump is in service
- 2LT-112, VCT level, transmitter fails HIGH

Which of the following describes the operational impact to the CVCS system and the effect on the running charging pump, if any (assume no operator action taken)?

A. No Auto make-up, VCT level lowers, Auto swap to RWST, and no effect on 21 charging pump.

B. No Auto make-up, VCT level lowers, no Auto swap to RWST, subsequent loss of suction and damage to 21 charging pump.

C. Auto make-up maintains VCT level above the low-low level swapover; no effect on 21 charging pump.

D. Auto make-up initiates but unable to maintain level in VCT, Auto swap to RWST; no effect on 21 charging pump.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly recalls that auto swapover will actuate when 1 out of 2 VCT levels are low-low. Incorrect in that auto swapover requires 2 out of 2 VCT levels low-low..

B. Correct. 2LT-112 failing high will cause the CV35 high level divert valve to open. This will result in actual VCT level lowering. With the LT-112 failed high, auto make-up will not actuate and auto swapover to the RWST on VCT low-low level will not function due to needing 2 out of 2 VCT levels low-low to initiate auto swapover. As VCT continues to lower, the inservice centrifugal charging pump will lose suction and eventually run drying causing pump damage.

C. Incorrect. Plausible if the operator believes that 2LT-114 will also actuate auto makeup. Incorrect in that only 2LT-112 will actuate auto make-up when VCT low level is reached.

D. Incorrect. Plausible if the operator believes that 2LT-114 will also actuate auto make-up and that 1 out 2 VCT levels will cause auto swapover. Incorrect in that only 2LT-112 actuates auto make-up and auto swapover requires 2 out of 2 VCT levels low-low.

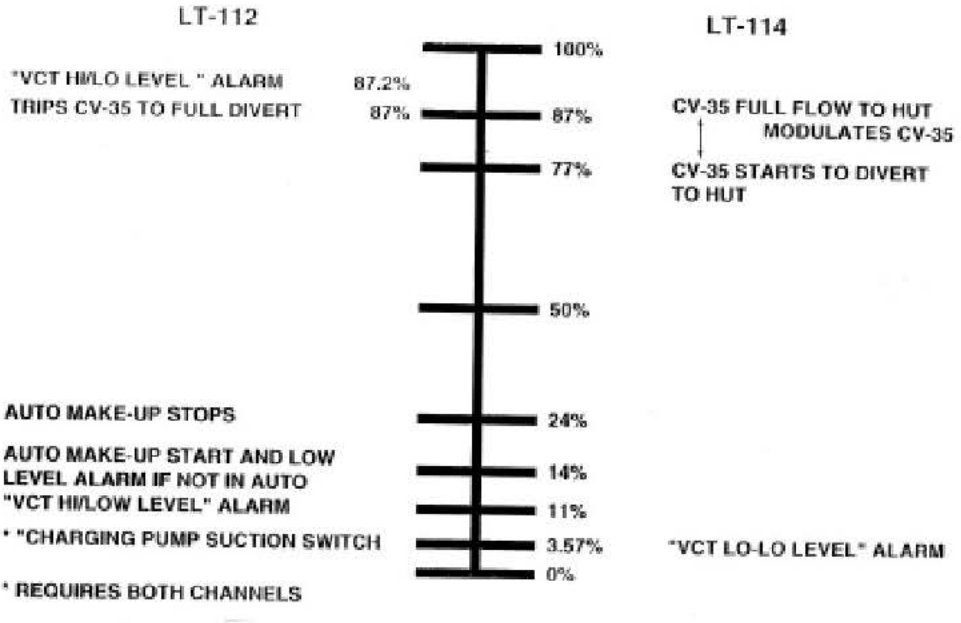
Technical References:	NOS05CVCS00-18
Proposed References to be provided:	None
Learning Objective:	NOS05CVCS00-18, Objective 15. Given plant conditions, relate the Chemical and Volume Control System and the following: Emergency Core Cooling System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	02-CVCS
K/A #:	004 K3.13 Knowledge of the effect that a loss or malfunction of the CVCS system will have on the following systems or system parameters: ECCS.
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how a failure of a VCT level transmitter will affect the in-service charging pump.

Additional Reference:

S2.OP-AB.CVC-0001(Q)

EXHIBIT 2
VCT LEVEL CONTROLLER FUNCTIONS



30 **04-072** **Points: 1.00**

Given:

- Salem Unit 1 is in MODE 4 performing a shutdown.
- 11 RHR Pump is in service
- 11 RHR Heat Exchanger is in service
- 11RH18 and 12RH18 are being throttled
- 1RH20 is being throttled
- The BOP operator has determined that the cooldown is approaching the maximum allowable Technical Specification limit

Which of the following describes the actions that the operator will take to LOWER the cooldown rate while maintaining a constant RHR flow rate?

A. Lower the demand on the RH18s and Raise the demand on the 1RH20.

B. Raise the demand on the RH18s and Lower the demand on the 1RH20.

C. Lower the demand on the RH18s and Lower the demand on the 1RH20.

D. Raise the demand on the RH18s and Raise the demand on the 1RH20.

Answer: A

Explanation / Justification:

A. Correct. In order to lower the cooldown rate and maintain current flowrate thru the 11 RHR HX, the operator will need to lower the demands on the RH18s to close them more to reduce flow thru the 11 RHR HX and raise the demand on 1RH20 to raise the bypass flow around the 11 RHR HX.

B. Incorrect. Plausible if the operator believes that raising demand on the RH18s will close the valves and lowering demand on the RH20 will open the valve. Incorrect in that raising demand on the RH18s opens them more and lowering demand on the RH20 will close the valve resulting in raising the cooldown rate.

C. Incorrect. Plausible because lowering the demands on the RH18s and the RH20 will lower the cooldown rate but will also lower the overall RHR flowrate.

D. Incorrect. Plausible if the operator believes raising the demands on the RH18s and the RH20 will lower the cooldown rate. Incorrect in that raising the demands will open the RH18s more and the RH20 more resulting in rise in cooldown or possible not effect on the cooldown rate.

Technical References:	NOS05RHR000-19
Proposed References to be provided:	None
Learning Objective:	NOS05RHR000-19, Objective 4. Describe how the following components impact the Residual Heat Removal System during normal and abnormal conditions: RH18s and RH20
Question Source:	Modified – Salem 2020 NRC RO31
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	04P-RHR System
K/A #:	005 A4.02 Ability to manually operate and/or monitor the RHR System in the control room: RHR heat exchanger temperature/bypass control valves
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how to manually operate the RH18s and RH20 (RHR HX bypass valve) to control the cooldown rate.

Parent Question: Salem 2020 NRC RO31

Modified question based on the stem asking about the cooldown.

Exam Outline Cross Reference:	Level:	RO SRO
	Tier #	2
	Group #	1
	K/A #	005000K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.
Importance:		2.5

Question: #31

Given:

- Unit 2 is in MODE 5.
- 21 RHR HX Loop is providing Shutdown Cooling with a cooldown rate of 20 °F/hr.
- 21RH18, RHR HX Flow Control Valve, is throttled at 40 % to maintain total flow at 2500 gpm.
- 2RH20, RHR HX Bypass Valve, is throttled at 30 % open.

Subsequently the following occurs:

- Foreign material dropped in the Refueling Cavity is introduced into the RHR system causing partial blockage on the 21 RHR HX tubes.
- Cooldown rate has lowered to 5 °F/hr.

Based on the above conditions, in order to maintain a 20 °F/hr cooldown rate, the operator will be required to _____. (Assume decay heat load remains constant)

A. Lower the demand on the 21RH18 and Lower the demand on the 2RH20.

B. Lower the demand on the 21RH18 and Raise the demand on the 2RH20.

C. Raise the demand on the 21RH18 and Raise the demand on the 2RH20.

D. Raise the demand on the 21RH18 and Lower the demand on the 2RH20.

Answer: D

31 **02-032** **Points: 1.00**

Which of the following RCS break locations results in the highest peak clad temperature during a small break LOCA?

A. Cold Leg

B. Hot Leg

C. CRDM on Reactor Head

D. Pressurizer Steam Space

Answer: A

Explanation / Justification:

A. Correct. Due to thermodynamics, a cold leg break gives the highest peak clad temperatures. Also, ECCS pumps inject into the cold legs.

B. Incorrect. Plausible because a hot leg break will result in the highest loss of flow. Incorrect in that this accident does not result in the highest peak clad temperatures.

C. Incorrect. Plausible because a Control Rod Ejection is an analyzed SBLOCA. Incorrect in that this accident does not result in the highest peak clad temperatures.

D. Incorrect. Plausible because vapor space accidents at the PZR is an accident analyzed and located on the hot leg. Incorrect in that this accident does not result in the highest peak clad temperatures.

Technical References:	NOS05ECCS00-18
Proposed References to be provided:	None
Learning Objective:	NOS05ECCS00-18, Objective 1. Describe the purpose of the Emergency Core Cooling System.
Question Source:	Bank – Braidwood 2018 RO41
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	02-Emergency Core Cooling System
K/A #:	006 K5.07 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to ECCS: Expected temperature values in various locations of the RCS due to different break locations during all ECCS injection modes
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how what RCS break location represents the worst case scenario from an accident analysis standpoint.

32 **05-021** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Console Alarm PRT LEVEL HI-LO is LIT
- PRT level is 90%

Which of the following completes the statement?

PRT level will be lowered…

A. via RCDT Pump to the Auxiliary Building Sump Tank

B. via RCDT Pump to a CVCS Hold-Up Tank

C. by gravity drain to a CVCS Hold-Up Tank

D. by gravity drain to the Auxiliary Building Sump Tank

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because most drain in the Aux Building drain to the Aux. Bldg. Sump Tank. Incorrect in that the PRT does not drain to the Aux. Bldg. Sump Tank.

B. Correct. To lower level in the PRT, S2.OP-SO.PZR-0003 will drain the PRT by opening the 2PR14 which will automatically open the 2WL12 which starts a RCDT Pump and defeats the low level cu-off. The RDCT pumps can discharge to either (1) RWST, (2), CVCS HUTs, or (3) WHUTs.

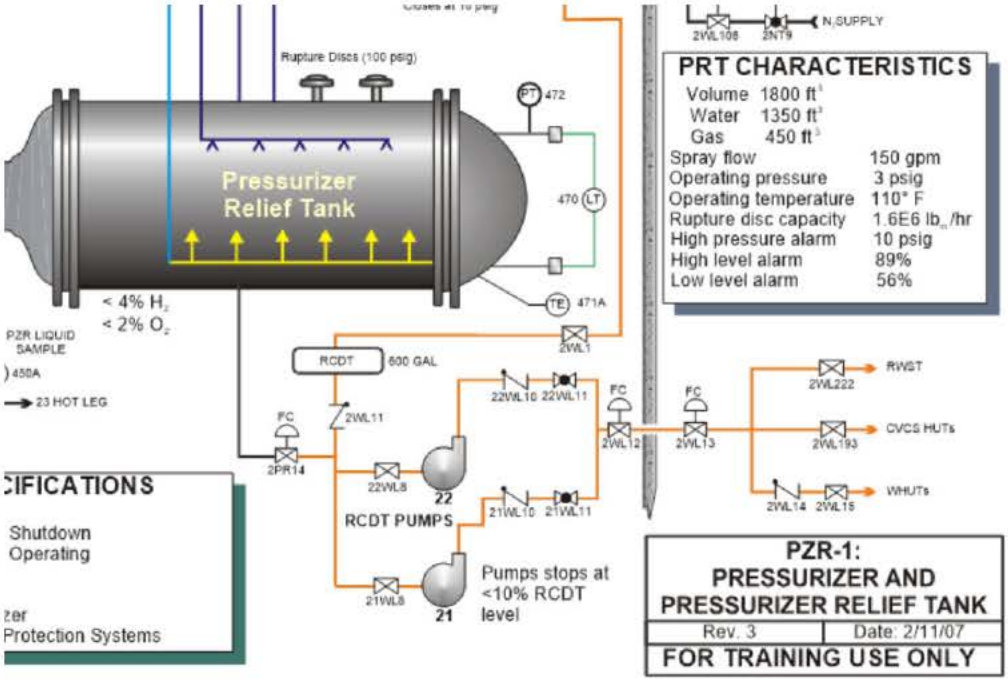
C. Incorrect. Plausible because the operator may believe that the RCDT Pumps are only used to drain down the RCDT and that gravity drain is the method used to drain down the PRT.

D. Incorrect. Plausible because the operator may believe that the RCDT Pumps are only used to drain down the RCDT and that gravity drain is the method used to drain down the PRT.

Technical References:	S2.OP-SO.PZR-0003 (R17), Simplified PZR-1
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPRT-09, Objective 4.g. Describe the function of the following components and how their normal and abnormal operation affects the Pressurizer and Pressurizer Relief Tank: Pressurizer Relief Tank
Question Source:	New
Question Cognitive Level:	Fundamental/Analysis
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	05-Pressurizer Relief/Quench Tank System
K/A #:	007 K1.06 Knowledge of the physical connections and/or cause and effect relationships between the PRT System and the following systems: LRS
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the PRT system is physically connected to the LRS. In this case, the PRT is drained to LRS via CVCS Hold-up Tanks or Waste Hold-up Tanks.

Additional Reference:



33 **08-021** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA and a Loss of Off-site Power
- The crew is implementing 2-EOP-TRIP-1, Reactor Trip or Safety Injection

Which of the following describes the operational impact on the Component Cooling Water System (CCW) and the reason why?

A. The CCW Pumps previously running remain running and both CCW heat exchangers are available; no likelihood for SW pump runout.

B. NO CCW Pumps are running and Service Water (SW) to the CCW heat exchangers is automatically isolated to prevent SW pump runout and diesel generator overloading.

C. NO CCW Pumps are running and SW to the CCW heat exchangers must be manually isolated to prevent SW pump runout and diesel generator overloading.

D. ALL CCW Pumps will be running and SW to the CCW heat exchangers is isolated to prevent excessive SW pump runout.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because this is how the CCW system operates when the SECs are operating in Mode 1 condition (SI only). Incorrect on that the SECs are in Mode 3 (SI plus blackout).

B. Correct. The plant is in SEC Mode 3 (SI plus Blackout) condition which will trip all CCW Pumps and isolate (close) service water flow to both CCW heat exchanger inlet valves. The SW is isolated to prevent runout conditions with only three SW pumps running and the CCW Pumps do not get sequenced onto the diesel due to loading concerns on the diesel generator. A CCW Pump is started per 2-EOP-APPX-1 to support safeguard loads and RHR HX operation during Cold Leg recirculation operation.

C. Incorrect. Plausible because no CCW pumps are running during a SEC Mode 3 condition and SW is required to be isolated to prevent SW pump runout. Incorrect in that the SW flow to the CCW HXs are automatically isolated during an SEC Mode 3 condition.

D. Incorrect. Plausible because this is how the CCW system operates when the SECs are operating in Mode 2 condition (Blackout only). Incorrect on that the SECs are in Mode 3 (SI plus blackout).

Technical References:	NOS05CCW000-15
Proposed References to be provided:	None
Learning Objective:	NOS05CCW000-15, Objective 16.e. Given plant conditions, relate the Component Cooling Water System with the following: Safeguards Equipment Controller (SEC)
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	08-Component Cooling Water System
K/A #:	008 K5.10 Knowledge of the operational implications of the following concepts as they apply to the CCWS: Requirements on and for the CCWS for different conditions of the power plant
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the operational impact on the CCW Pumps during an Accident plus Blackout plant condition and the reason why.

34 **03-026** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Master Pressure Controller (MPC) is in AUTO
- PZR Pressure Channel I is selected for Control
- PZR Pressure Channel IV is selected for Alarm

Then:

- A malfunction with the MPC causes the output demand to fail LOW (0%)

Which of the following identifies the effect on RCS pressure and the expected system response, assuming no operator action?

RCS pressure will…

A. lower and NO PZR Backup Heaters will energize.

B. lower and BOTH PZR Backup Heaters will energize.

C. rise and NO PZR PORVs will open.

D. rise and BOTH PZR PORVs will open.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall how the MPC demand effects the operation of the PZR heaters and spray valves. Incorrect in that a 0% demand from the MPC will energize both PZR heater backup groups and cause RCS to rise.

B. Incorrect. Plausible because the operator may incorrectly recall how the MPC demand effects the operation of the PZR heaters and spray valves. Incorrect in that a 0% demand from the MPC will energize both PZR heater backup groups and cause RCS to rise.

C. Incorrect. Plausible because the operator may incorrectly interpret how the MPC demand effects the operation of the PZR PORVs and believe with the demand failed low no PZR PORVs will open. The figure in S2.OP-AB.PZR-0001 Attachment 2 can mislead the operator thinking that the MPC does have some control function on the PZR PORVs. Incorrect in that the MPC does not control the PZR PORVs.

D. Correct. When the Master Pressure Controller (MPC) output fails low (0% demand), spray valves will close and both PZR Backup Heaters will energize raising RCS pressure. As RCS pressures rises, PZR sprays will not open since they are controlled from the MPC. Both PZR PORVs will open when RCS pressure exceeds 2335 psig.

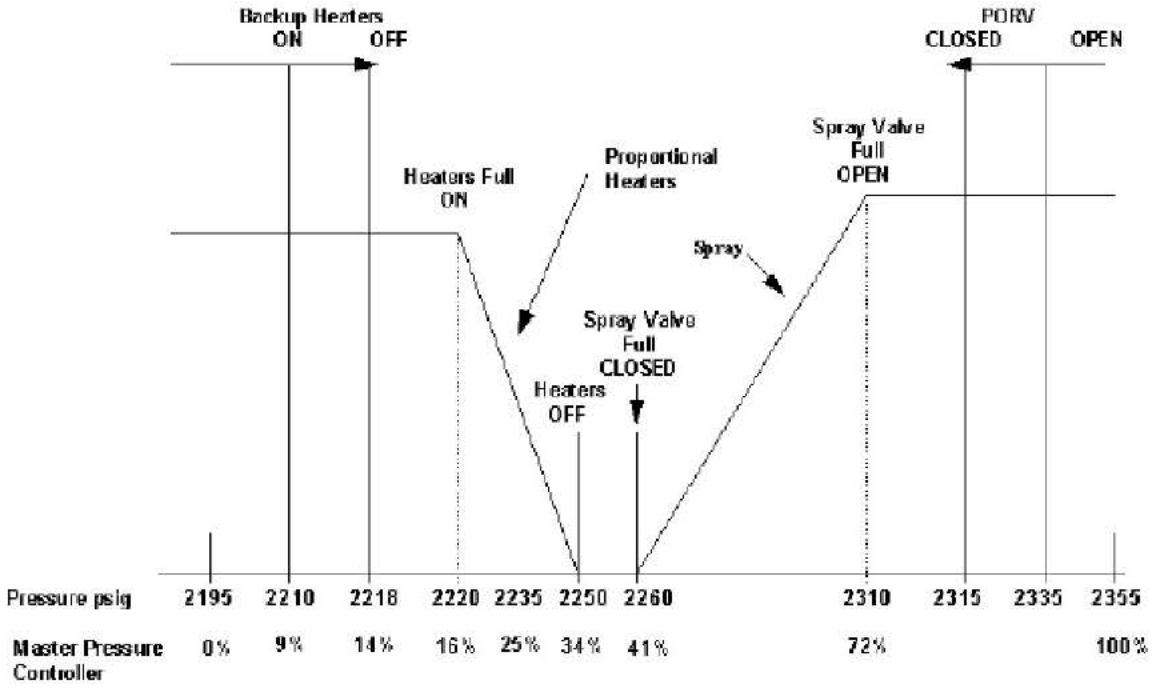
Technical References:	NOS05PZRPL-12, S2.OP-AB.PZR-0001 (R20)
Proposed References to be provided:	None
Learning Objective:	NOS05PZRPL-12, Objective 9. State the setpoints, coincidence, blocks and permissives for automatic actuations associated with the Pressurizer Pressure and Level Control System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	03-Pressurizer Pressure Control System
K/A #:	010 K4.03 Knowledge of PZR Pressure Control System design features and/or interlocks that provide for the following: Overpressure control
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the design feature with the operation of the PZR PORVs for overpressure control. Specifically, the candidates is being tested on how a Master Pressure Controller fails low will effect RCS pressure and whether the PZR PORVs will open. Some plant designs operate the PORCs from the MPC controller. Salem's design the PZR PORVs are interlocked directly from the PZR pressure channels..

Additional Reference: S2.OP-AB.PZR-0001

ATTACHMENT 2
(Page 1 of 1)

PRESSURIZER PRESSURE CONTROL



35 **07-024** **Points: 1.00**

Given:

- Salem Unit 2 is at 7% power

Which of the following conditions would DIRECTLY result in a reactor trip?

A. Three FLOW LO bistables tripped on one RC Loop.

B. Three PZR PRESS LO bistables tripped.

C. Two PZR PRESS HI bistables tripped.

D. One SG LO LEVEL bistable tripped on two SGs.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the reactor trip logic is met based on 2/3 flows < 90% on 1/4 RC Loops. Incorrect in that this reactor trip is enabled when reactor power is above P-8 (36%). Note, above P-7 requires 2/4 RC loop flows < 90%.

B. Incorrect. Plausible because the reactor trip logic is met based on 2/4 PZR pressure < 1865 psig. Incorrect in that this reactor trip is blocked until reactor power is above P-7 (10%).

C. Correct. PZR Pressure High on 2/3 channels > 2385 psig will cause a reactor trip. This trip is not blocked in any plant condition.

D. Incorrect. Plausible because the operator may confuse the trip logic as 1/3 NR level low on 1/4 SGs. Incorrect in that reactor trip requires 2/3 NR levels low on 1/4 SGs.

Technical References:	Fluency List
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank – Point Beach 2017 RO1
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	07-Reactor Protection System
K/A #:	012 A3.06 Ability to monitor automatic features of the RPS including: Trip logic
Importance:	4.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked to monitor the given plant conditions to determine if an automatic reactor tripped has occurred.

Additional Reference:

VI. REACTOR TRIP SETPOINTS

SLIDE-17; ELO-12 (Operator Fundamental – Closely Monitor Plant Indications)

Name	Setpoint	Coincidence	Permissive
Manual	Manual	1/2 Switches	None
SI	Any SI	1 key switch ("S" signal) on either train	None
SR High Flux	1.0×10^5 cps	1/2 SR Detectors	P-6 allows block
IR High Flux	25 %	1/2 IR Detectors	P-10 allows block
PR High Flux LO SP	25%	2/4 PR Detectors	P-10 allows block
PR High Flux Hi SP	109%	2/4 PR Detectors	None
PR High Positive Rate	+5 % of RTP w/2 sec TC	2/4 PR Detectors	None
OTΔT	Variable[1]	2/4 ΔT's	None
OPΔT	Variable[2]	2/4 ΔT's	None
SG LO-LO Level	14 %	2/3 levels on 1/4 SG's	None
RCP Breaker Open	2 Breakers Open	2 RCP breakers	> P-7
Loss of Flow	<90%	2/3 on 1/4 RC Loops	> P-8
Loss of Flow	<90%	2/3 on 2/4 RC Loops	> P-7
RCP Undervoltage	<70% for >0.1 sec	1/2 on Group Buses H or E (AND) 1/2 on Group Buses F or G	> P-7
RCP Underfrequency	<56.5Hz, >0.1 sec	Same as above	> P-7
PZR Pressure High	>2385 psig	2/4 PZR Press Channels	None
PZR Pressure Low	<1865 psig	2/4 PZR Press Channels (rate compensated)	> P-7
PZR Level High	>92%	2/3 PZR Level Channels	> P-7
Turbine Trip	<45 psig ASOP[3] or 4/4 stop valves closed	2/3 channels for <45 psig or Stop valves 15% closed from open limit	> P-9

[1] $OT\Delta T_{setpoint} = \Delta T_0 \{ K1 - K_2 [(1 + \tau_1 s) \div (1 + \tau_2 s)] (T - T') + K_3 (P - P') - f_1(\Delta\phi)\}$

[2] $OP\,\Delta T_{setpoint} = \Delta T_0 \{K_4 - K_5 [(\tau_3 s \div (1 + \tau_3 s)] T - K_6 (T - T'') - f_2(\Delta\phi)\}$

36 **07-025** **Points: 1.00**

Given:

- Salem Unit 2 is at 18% Power
- 21 and 22 SPTs are supplying the 4KV Group Buses

Then:

- An electrical transient occurs on the 500 KV grid.
- During the transient the following values were reached:

4KV Group Bus	H	E	F	G
Frequency (Hz)	56	56	59.7	59.7
Control Room Indication (VAC)	4100	4100	4200	4200

Which ONE of the following describes the expected status of the reactor trip breakers and the Reactor Coolant Pumps (RCPs) assuming no operator action?

A. Reactor Trip breakers remain closed and ALL RCPs remain running.

B. Reactor Trip breakers remain closed and ONLY 2 RCPs remain running.

C. Reactor Trip breakers open and ALL 4 RCPs trip off.

D. Reactor Trip breakers open and ONLY 2 RCPs remain running.

Answer: A

Explanation / Justification:

A. Correct. The reactor trip logic from an RCP UF/UV is based on 1 out of 2 buses taken twice at 56.5 Hz (H bus or E bus AND F bus or G bus) trips all four RCPs and if above P-7 (10% power), initiates a reactor trip. In this case, the reactor trip logic is not satisfied since the underfrequency condition is only on H bus and E bus. The individual 4KV breakers for the RCP motors will trip open on undervoltage conditions and not on undefrequency. So in this, the crew would receive OHA J-37, 4KV GRP BUS UNDRFREQ, and all RCPs will remain running with the reactor trip breakers closed.

B. Incorrect. First part is correct. Second part is incorrect. Plausible if the operator makes an incorrect assumption that the 4KV Group Bus breakers open on underfrequency conditions. This is not true, the RCP breakers open on an undervoltage condition.

C. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that the underfrequency on H and E Bus will trip open 21 and 22 RCP breakers, thus satisfying the 2/4 RC loop low flow reactor trip logic with the plant below P-8 (< 36%) but above P-7 (>10%). Second part is incorrect. Plausible because if the operator believes that the UF condition on H and E Buses satisfies the 1 out of two taken twice, resulting in all RCPs tripping.

D. Incorrect. First part is incorrect. Plausible because the operator may incorrectly believe that the underfrequency on H and E Bus will trip open 21 and 22 RCP breakers, thus satisfying the 2/4 RC loop low flow reactor trip logic with the plant below P-8 (< 36%) but above P-7 (>10%). Second part is incorrect. Plausible if the operator believes that only the RCP breakers on H and E Bus will trip open on a UF condition, which is not true.

Technical References:	Fluency List
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	07-Reactor Protection System
K/A #:	012 K1.09 Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: RCPS
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the RCP system relates to the RPS system. In this case, the stem provides conditions (underfrequency and undervoltage) to determine if the applicant knows the trip logic associated with RCP underfrequency.

Additional Reference:

VI. REACTOR TRIP SETPOINTS

SLIDE-17; ELO-12 (Operator Fundamental – Closely Monitor Plant Indications)

Name	Setpoint	Coincidence	Permissive
Manual	Manual	1/2 Switches	None
SI	Any SI	1 key switch ("S" signal) on either train	None
SR High Flux	1.0×10^5 cps	1/2 SR Detectors	P-6 allows block
IR High Flux	25 %	1/2 IR Detectors	P-10 allows block
PR High Flux LO SP	25%	2/4 PR Detectors	P-10 allows block
PR High Flux Hi SP	109%	2/4 PR Detectors	None
PR High Positive Rate	+5 % of RTP w/2 sec TC	2/4 PR Detectors	None
OTΔT	Variable[1]	2/4 ΔT's	None
OPΔT	Variable[2]	2/4 ΔT's	None
SG LO-LO Level	14 %	2/3 levels on 1/4 SG's	None
RCP Breaker Open	2 Breakers Open	2 RCP breakers	> P-7
Loss of Flow	<90%	2/3 on 1/4 RC Loops	> P-8
Loss of Flow	<90%	2/3 on 2/4 RC Loops	> P-7
RCP Undervoltage	<70% for >0.1 sec	1/2 on Group Buses H or E (AND) 1/2 on Group Buses F or G	> P-7
RCP Underfrequency	<56.5Hz, >0.1 sec	Same as above	> P-7
PZR Pressure High	>2385 psig	2/4 PZR Press.Channels	None
PZR Pressure Low	<1865 psig	2/4 PZR Press.Channels (rate compensated)	> P-7
PZR Level High	>92%	2/3 PZR Level Channels	> P-7
Turbine Trip	<45 psig ASOP[3] or 4/4 stop valves closed	2/3 channels for <45 psig or Stop valves 15% closed from open limit	> P-9

[1] $OT\Delta T_{setpoint} = \Delta T_0 \{ K1 - K_2 [(1 + \tau_1 s) \div (1 + \tau_2 s)] (T - T') + K_3 (P - P') - f_1(\Delta\phi)\}$

[2] $OP\ \Delta T_{setpoint} = \Delta T_0 \{K_4 - K_5 [(\tau_3 s \div (1 + \tau_3 s)] T - K_6 (T - T'') - f_2(\Delta\phi)\}$

37 **02-033** **Points: 1.00**

Given:

- Salem Unit 1 experienced a reactor trip and an SI actuation due to a LOCA
- The crew is performing 1-EOP-LOCA-1, Loss of Reactor or Secondary Coolant
- SI has been RESET
- SECs have been RESET
- ALL EDGs have been stopped from the control room

Then:

- Loss of Off-site Power occurs

Which of the following describes the expected response of the EDGs and **safeguard loads**?

A. EDGs require manual start and some safeguard loads need to be manually started.

B. EDGs require manual start and SECs automatically sequence ALL safeguard loads.

C. EDGs automatically start and SECs automatically sequence ALL safeguard loads.

D. EDGs automatically start and some safeguard loads need to be manually started.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that since the EDGs were stopped and the SECs reset, that the EDGs will not auto start.

B. Incorrect. Plausible because the SEC will automatically sequence loads. Incorrect in that the SEC in Mode 2 will only start the charging pumps, the other safeguard loads need to be started manually.

C. Incorrect. Plausible because the SEC will automatically sequence loads. Incorrect in that the SEC in Mode 2 will only start the charging pumps, the other safeguard loads need to be started manually.

D. Correct. After the SI is reset and SECs are reset, the SI signal to the SEC logic are cleared and the SECs are ready for automatic actuation. The EDGs will automatically start on a Blackout condition (Mode 2) and the SECs will automatically sequence loads for Blackout only. The only safeguards load started during Mode 2 are the charging pumps. The crew will be required to start the other safeguard loads (e.g, SI pumps, RHR Pumps, and CFCUs) since the SI and SECs were reset prior to the LOOP.

Technical References:	NOS05SEC000-09
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 4. Describe in detail each Mode of Safeguards Equipment Control System operation, including setpoints for automatic actuation
Question Source:	Bank – Commanche Peak 2020 RO20
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	02-Engineered Safety Features Actuation System
K/A #:	013 A1.14 Ability to predict and/or monitor changes in parameters associated with operating the ESFAS system including: EDG
Importance:	3.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict how the EDGs will respond to a LOOP after the SI signal and SECs have been reset.

38 **05-022** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21, 22, 23 and 24 CFCUs are running in HIGH speed with 25 CFCU in standby
- An inadvertent SI actuation occurs

Which of the following completes both statements?

1. Following SI actuation, all CFCUs will start in LOW speed __(1)__.
2. To restore CFCUs to HIGH speed operation, __(2)__ must be RESET.

A. (1) after 20 seconds
(2) Only SI

B. (1) after 20 seconds
(2) SI and SECs

C. (1) immediately
(2) SI and SECs

D. (1) immediately
(2) Only SI

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may believe that with an SI only, that only resetting is required. Incorrect in that the SECs are required to be reset as well.

B. Correct. Inadvertent SI actuation will actuate the SECs in MODE 1 operation. The CFCUs running in high speed will be tripped and started 20 seconds later in low speed. The CFCU in standby will start 20 seconds later following the SI actuation. In order to regain manual control of the CFCUs to return them to high speed operation, the operator will need to reset both trains of SI and reset all SECs. Step 1 of 2-EOP-TRIP-3, SI Termination, will accomplish this.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that with no loss of off-site power present, then the CFCUs can immediately start since no diesel overloading issue. Incorrect in that a time delay is still required to allow the CFCUs to coast down. Part 2 is correct.

D. Incorrect. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that with no loss of off-site power present, then the CFCUs can immediately start since no diesel overloading issue. Incorrect in that a time delay is still required to allow the CFCUs to coast down. Part 2 is incorrect. Plausible because the operator may believe that with an SI only, that only resetting is required. Incorrect in that the SECs are required to be reset as well.

Technical References:	NOS05SEC000-09, 2-EOP-TRIP-3 Sht. 1(R41)
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 9. Describe how to reset the Safeguards Equipment Control System after a Mode Operation, and what happens when the Safeguards Equipment Control System is reset
Question Source:	Modified – Braidwood 2016 NRC RO20
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	05-Containment Cooling System
K/A #:	022 A2.07 Ability to (a) predict the impacts of the following on the Containment Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: ESFAS actuation
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to how the ESFAS actuation affects the status of the CFCUs and what action(s) is necessary to regain control of the CFCU to restore to normal operation.

Parent Question: Braidwood 2016 NRC RO20

Modified question based on adding Part 2 to ask what actions are required to restore CFCUs to high speed operation.

20 **ID: 1267989** **Points: 1.00**

Given:

- Unit 1 is at full power.
- 1A, 1B and 1C RCFCs are running in high speed with the 1D RCFC in standby.
- A reactor trip and Safety Injection occur due to high containment pressure.

After the SI occurs, what is the response of the RCFCs?

A. ALL RCFCs start immediately in low speed.

B. 1A, 1B and 1C RCFCs start immediately in low speed, and the 1D RCFC starts 20 seconds later in low speed.

C. 1A, 1B and 1C RCFCs start 20 seconds later in low speed, and the 1D RCFC starts immediately in low speed.

D. ALL RCFCs start 20 seconds later in low speed.

Answer: D

39 **05-023** **Points: 1.00**

Given:

- Salem Unit 2 experienced a reactor trip and safety injection actuation due to a LOCA
- 23 CFCU is C/T for maintenance
- 2B 4KV Vital Bus de-energized on Bus Differential protection

Which of the following identifies the status of the CFCUs?

A. 22 and 25 are running Only.

B. 22 and 24 are running Only.

C. 21 and 24 are running Only.

D. 21 and 25 are running Only.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 23 and 25 CFCUs are supplied from A and C bus. Incorrect in that only 22 and 24 CFCUs will not be running due to B bus de-energized.

B. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 22 and 24 CFCUs are supplied from A and C bus. Incorrect in that 22 and 24 CFCUs will not be running due to B bus de-energized.

C. Incorrect. Plausible if the operator incorrectly recalls the power supplies to the CFCUs and believes that 21 is supplied from A bus and 22 and 24 CFCUs are supplied from C bus. Incorrect in that 22 and 24 CFCUs will not be running due to B bus de-energized

D. Correct. The CFCUs are supplied from the following 4KV Vital Busses; 21 – A bus, 22 – B bus, 23 – C bus, 24 – B bus, and 25 – C bus. With 2B 4KV bus de-energized and 23 CFCU C/T, then only 21 and 25 CFCUs are running.

Technical References:	NOS05CONTMT-17
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 4. State the power supply to the following Containment and Containment Support Systems components, including voltage level and 1E/Non 1E
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment Cooling System
K/A #:	022 K2.01 Knowledge of the electrical power supplies to the following: Containment Cooling System CCS fans
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked which CFCUs will be running following a loss of 2B 4KV bus.

40 **05-024** **Points: 1.00**

Given:

- Salem Unit 2 experienced a LOCA
- Containment pressure is 10 psig and slowly rising
- ALL available safeguards equipment have started with the exception that 2B SEC failed to actuate
- SECs have NOT been RESET

Based on the conditions above, which of the following completes the statement?

__(1)__ containment spray pump(s) is/are still available from a Class 1E vital power source and __(2)__ automatically start when containment pressure exceeds the containment pressure HI-HI setpoint.

A. (1) ONLY one (2) will

B. (1) ONLY one (2) will NOT

C. (1) BOTH (2) will

D. (1) BOTH (2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible if the operator incorrectly recalls the power supplies to the CS pumps and believes that when the SEC is at the end of the sequence the CS pumps will not start when a containment pressure HI-HI signal is present. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present. Part 2 is correct.

B. Incorrect. Part 1 and Part 2 are incorrect. Plausible if the operator incorrectly recalls the power supplies to the CS pumps and believes that when the SEC is at the end of the sequence the CS pumps will not start when a containment pressure HI-HI signal is present. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present.

C. Correct. Both containment spray pumps will still be powered from the 2A and 2C 4KV Vital Busses with the 2B SEC failing to actuate. The 2A and 2C SECs have actuated in MODE 1 due to a LOCA only and with the sequence complete and the SECs not reset, a containment pressure HI-HI signal will automatically start the 21 and 22 CS pumps since the SECs are still waiting for a CS HI-HI signal when sequence is complete.

D. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the SECs sequencer allows for two opportunities to start a CS pump. The operator may incorrectly recall this and believe that when the SEC sequence is complete, the operator will have to manually start a CS pump. Incorrect in that at the end of sequence and with the SEC not reset, the SEC will auto start a CS pump when a containment pressure HI-HI signal is present.

Technical References:	NOS05SEC000-09
Proposed References to be provided:	None
Learning Objective:	NOS05SEC000-09, Objective 4. Describe in detail each Mode of Safeguards Equipment Control System operation, including setpoints for automatic actuation
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment Spray System
K/A #:	026 K2.01 Knowledge of electrical power supplies to the following: Containment Spray System containment spray pumps
Importance:	3.9
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked which containment spray pumps are still available to start given a loss of the 2B SEC and will the containment spray pumps auto start when the containment spray HI-HI signal is exceeded.

Additional References: NOS05SEC000

LESSON NAME:	SAFEGUARDS EQUIPMENT CONTROL SYSTEM	
	NOS05SEC000-09	02/18/21

g. Containment Spray Pump Sequencing

1) If the SSPS Containment HI-HI Pressure signal is not present when the SEC initially tries to start the Spray Pumps, the SEC contact will re-open

2) Once the SEC has completed the last step of its loading sequence, the CS Pump start contact is re-closed

a) If Hi-Hi Containment Pressure occurs the CS Pumps will auto start

b) If the SEC has been reset, the CS Pumps will NOT respond to a HI-HI Containment Pressure until the SEC is again actuated

c) CS Pumps do not have circuit breaker "a" contacts in the "Sequence Failure" or "Loading Complete" circuits. Sequence Failure will NOT occur as a result of a CS Pump not starting until the SSPS HI-HI Pressure signal is received. CS pumps also do not have individual component status lights on RP-4 Status Panel

41 **04-073** **Points: 1.00**

Given:

- Salem Unit 2 reactor tripped and actuation of Main Steam Line Isolation due to a steamline break
- ONLY the 23 AFW Turbine-Driven feedpump is running supplying all SGs
- ALL SG pressures are 900 psig and lowering rapidly

Which of the following completes the statement concerning the effect on the AFW system (assuming no operator action)?

When SG pressures lowers to 700 psig, 23 AFW turbine speed will __(1)__ and feedflow will ___(2)___.

A. (1) lower (2) lower

B. (1) remain constant (2) rise

C. (1) remain constant (2) remain constant

D. (1) lower (2) rise

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that as SG pressure lowers so will the turbine speed and feedflow.

B. Correct. 23 AFW Turbine Driven Pump is designed to operate with SG pressures at 100 psig. As SG pressures lower, the 23 AFW governor will throttle to maintain relative speed of the turbine constant. However, as SG pressure lowers, the D/P across the feedpump discharge and the SG will increase resulting in more AFW flow to the SGs.

C. Incorrect. Plausible because turbine speed will be maintain relatively constant and the operator may believe that feedflow will also remain constant. Incorrect in that feedflow will rise due to the lowering SG pressure.

D. Incorrect. Plausible because feedflow will rise as SG pressure lowers.

Technical References:	NOS05AFW000-18
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 15.c. Given plant conditions, relate the Auxiliary Feedwater System with the following, Main Steam
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	04-Main and Reheat Steam System
K/A #:	039 K3.03 Knowledge of the effect that a loss or malfunction of the Main and Reheat Steam System will have on the following systems or system parameters: AFW system
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked knowledge of the effects of the turbine driven AFW pump when SG pressure lowers due to a faulted SGs and how feedflow will also behave with lowering SG pressure.

Additional Reference:

c. Turbine

1) Single stage, atmospheric condensing, Terry Turbine

a) Inlet steam conditions

(1) Normal - 1045 psig, 552°F

(2) Maximum - 1045 psig, 552°F

(3) Minimum - 85 psig, 327°F

42 **04-074** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21BF19, MFW Regulating Valve, fails fully OPEN

Which of the following completes the statement regarding how the plant will respond?

The reactor will trip and the crew will enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection, when high level on SG NR levels reach __(1)__ followed by a __(2)__.

	(1)	(2)
A.	67%	Feedwater Isolation
B.	67%	Feedwater Interlock ONLY
C.	92%	Feedwater Isolation
D.	92%	Feedwater Interlock ONLY

Answer: A

Explanation / Justification:

A. Correct. Failed open BF19 will result in a Main Turbine Trip on high SG level followed by a reactor trip. The SG High Level setpoint is 2 out of 3 SG NR levels ≥ 67% on any one SG (P-14). P-14 will trip the main turbine that will cause an auto reactor trip and actuate a FW Isolation to close all BF13s, BF19s, BF40s, trip both SGFPs, and trips Main Turbine. FW interlock only closes the BF19s and BF40s.

B. Incorrect. Plausible because a reactor trip will actuate a FW Interlock. Incorrect in that the SG high level (P-14) signal actuate a FW Isolation.

C. Incorrect. Plausible because this is the reactor trip setpoint for PZR high level.

D. Incorrect. Plausible because this is the reactor trip setpoint for PZR high level. Incorrect in that the P-14 setpoint is 67% and a FW Isolation occurs.

Technical References:	Drawing ESF-1
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objective 12.a. State the setpoints, coincidence, blocks and permissives for all Reactor Trips and Safety Injections actuations
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.8
Level:	RO
Category/System:	04-Mai Feedwater System / Emergency Procedures/Plan
K/A #:	059 G2.4.2 Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the setpoint for the SG High level Main Turbine/Reactor Trip and the auto action for FW Isolation.

43 **04-075** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Instrument air supply line to 2MS132, 23 AFW Turbine Steam Inlet Valve, ruptures

Which of the following completes both statements?

1. What is the status of 23 AFW Pump?
2. What action should the operator take FIRST?

A. (1) Running
(2) Reduce reactor power to prevent overpower condition.

B. (1) Running
(2) Take manual control of the Main Feedwater Regulating valves to prevent SG overfill.

C. (1) NOT Running
(2) Enter LCO 3.7.1.2, Auxiliary Feedwater System for INOPERABLE AFW Pump.

D. (1) NOT Running
(2) Fail open the 2MS132 by using the handhweel on the valve actuator.

Answer: A

Explanation / Justification:

A. Correct. Air supply line rupturing on the 2MS132 will cause the AFW steam inlet valve to fail open. This will result in 23 AFW running and injecting cold feedwater into the SGs. As a result, reactor power will increase and the operator should reduce reactor power..

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because with 23 AFW running additional feedwater is being supplied to all the SGs. Incorrect in that with digital feedwater system in Automatic, the MFW Regulating Valves will automatically adjust demand to maintain SG NR levels within program band.

C. Incorrect. Plausible because the operator may incorrectly recall how the valve fails on loss of air and believe the 2MS132 fails closed thereby preventing the start of 23 AFW pump. Entering TS LCO 3.7.1.2 is plausible because this LCO is applicable for the 23 AFW pump condition.

D. Incorrect. Plausible because the operator may incorrectly recall how the valve fails on loss of air and believe the 2MS132 fails closed thereby preventing the start of 23 AFW pump. Failing open the 2MS132 is plausible because many AOVs can be locally operated.

Technical References:	NOS05AFW000-18
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 16. Given an Auxiliary Feedwater System failure, predict the effect of the Auxiliary Feedwater System failure on the following: Steam Generators
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	04-Auxiliary Emergency Feedwater System
K/A #:	061 A2.07 Ability to (a) predict the impacts of the following on the Auxiliary Emergency Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Air operated valve, solenoid-operated valve, or motor-operated valve
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked how the AFW AOV will fail on loss of air and what action should the operator take based on the failure of the AOV.

44 **04-076** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The Primary Equipment Operator reports a significant air leak on the air supply line at the connection to the 2DR6, AFW Tank Makeup, valve actuator

Which of the following describes the expected impact to the Auxiliary Feedwater System (assuming the leak is causing a reduction in air pressure to the actuator) and operator response, if any?

AFW storage tank level will…

A. rise and overflow; local/manual control is NOT available.

B. rise and overflow; local/manual control is available.

C. remain the same; local/manual control is available.

D. remain the same; local/manual control is NOT available.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because not all air operated valves can be manually operated in the field.

B. Correct. The 2DR6 will fail to the open position on loss of air. This will result in the AFWST level rising and potentially overflowing the tank. Per S2.OP-AB.CA-0001, Attachment 2, the 2DR6 can be manually operated from the field to control tank level.

C. Incorrect. Plausible because the operator may believe that the 2DR6 failing open would result in tank overflow, thus its failure position would be closed to prevent this (air to open actuator).

D. Incorrect. Plausible because the operator may believe that the 2DR6 failing open would result in tank overflow, thus its failure position would be closed to prevent this (air to open actuator).

Technical References:	NOS05AFW000-18, S2.OP-AB.CA-0001 (R22)
Proposed References to be provided:	None
Learning Objective:	NOS05AFW000-18, Objective 15.f. Given plant conditions, relate the Auxiliary Feedwater System with the following: Demineralized Water System
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.3
Level:	RO
Category/System:	04-Auxiliary Feedwater System / Components
K/A #:	191001 K1.04 The failed-valve positions for different operators (open, closed, and as-is positions spring loaded valves; hydraulic, pneumatically controlled valves; electric motor-driven valves)
Importance:	3.2
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking how the loss of air to the 2DR6 AOV will be impacted and the effect on the AFWST level.

Additional Reference: S2.OP-AB.CA-0001, Attachment 2

AUXILIARY FEEDWATER SYSTEM

Valve	Fails	Description	Hdr/Bkup
21-24AF11	FO ♦	23AFP Disch to 21-24SG	2B/2A
21-24AF21	FO ♦	AFP to 21-24SG	2B/2A
21-23AF52	FC ♦	AFP Suct XConn	2B/2A
21-22AF40	FO ♦	21-22AF Pump Recirc	2B/2A
2MS53	FHSS ▽	23AFP Turbine Governor	2B/2A
2MS132	FO	23AFP Turbine Steam Inlet	2B/2A
2DR6	FO ♦	Supply to AFW	2B

♦ Can be manually operated

▽ Fails at High Speed Stop / Can be manually operated

45 **06-029** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 23 Charging Pump is in service
- 22 PZR Backup Heaters are ON
- A Overcurrent fault on the 2A 460V Motor Control Center (MCC) causes the 4160V feeder breaker 2A4D to trip open, de-energizing the 2A 460V and 230V MCCs

Per S2.OP-AB.460-0001, Loss of 2A 460 V Bus, which of the following describes the effect on plant operations and the required action(s)?

A. Loss of 2B1 125 VDC Battery Charger; place 2B2 125 VDC Battery Charger in service.

B. Loss of charging flow; start a centrifugal charging pump and restore PZR level.

C. Loss of all IRPI indications; transfer to emergency power source.

D. Loss of 22 PZR Backup Heaters; re-energize from emergency power bus.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall which 230 V bus supplies the normal and backup battery chargers and believe that the 2B1 125VDC is supplied from 2A 230 V bus. Incorrect in the2B1 125VDC Battery Charger is supplied from the 2B 230 V bus and the 2B2 125VDC battery charger is supplied from the 2A 230 V bus.

B. Correct. 23 charging pump is supplied from the 2A 460 V bus. An overcurrent condition on the 2A 460 V MCC/bus will cause the breakers associated on the 2A 460 and 230 V MCCs to trip open on overcurrent and de-energize the 2A 460V bus. This will result in loss of 23 charging pump and letdown isolation when the 23 charging pump breaker opens. Per S2.OP-AB.460-0001, Loss of 2A 460/230V Vital Bus, the operator will start a centrifugal charging pump and then re-establish letdown.

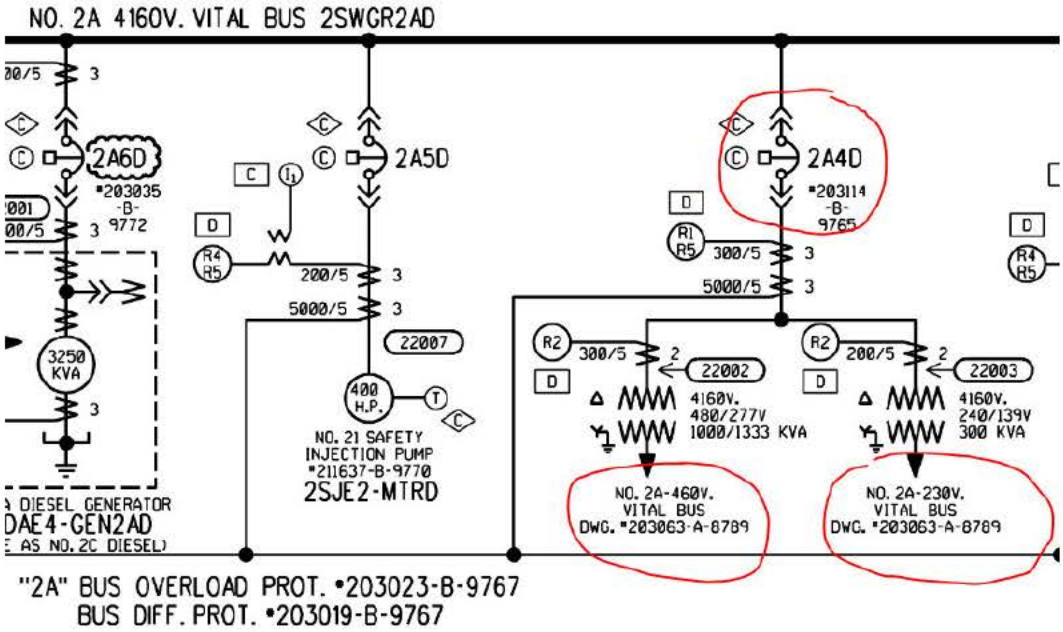
C. Incorrect. Plausible because the operator may incorrectly recall which 230 V bus supplies the normal and emergency power to the IRPI indications and believe that the 2A 230V bus is the normal supply resulting in a loss of IRPI indications. Incorrect in that the normal power supply is from 2B 460V and the 2A 230V bus supplies the emergency power.

D. Incorrect. Plausible because the operator may incorrectly recall which 460 V bus supplies the normal and emergency power to the Pressurizer backup heaters and believe that the 2A 460V bus is the normal supply resulting in a loss of PZR backup heaters Group 22. Incorrect in that the normal power supply is from the EP 460V bus and the 2A 460V bus supplies the emergency backup.

Technical References:	203061 (R39), 601390 (R25), S2.OP-AB.460-0001 (R9)
Proposed References to be provided:	None
Learning Objective:	NOS05460VAC-09, Objective 4.c. Describe the function of the following components and how their normal and abnormal operation affects the 460/230VAC Electrical System: 4KV/460V Group Transformers
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	06-AC Electrical Distribution System
K/A #:	062 A2.25 Ability to (a) predict the impacts of the following on the AC Electrical Distribution System and (b) based on those predications, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Fault on a motor control center.
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict the impacts on plant operations following an overcurrent condition on the 2A 460V bus that cause the upstream 4160V feeder breaker (2A4D) to trip open resulting in a loss of 2A 460 and 230V busses. The operator will recall knowledge of the major impacts from this loss on the major action from the abnormal procedure.

Additional References: 203061



46 **06-030** **Points: 1.00**

Given:

- Salem Unit 2 was at 100% power
- During control room log taking, the operator observes the 2A 125 VDC Bus ground ohmmeter reading 0K Ω (ohms)

Based on the above condition, which of the following completes both statements?

1. The reading indicates that a ground on the bus __(1)__ detected.
2. If a ground is detected, S2.OP-SO.125-0004, 125VDC Ground Detection, will __(2)__.

A. (1) is NOT
(2) open all breakers in the panel, then reclose them one at a time

B. (1) is NOT
(2) open one breaker at a time in the panel

C. (1) is
(2) open one breaker at a time in the panel

D. (1) is
(2) open all breakers in the panel, then reclose them one at a time

Answer: C

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground. Second part is incorrect. Plausible again if the operator confuses how to interpret resistance readings and believes a lower resistance is good..

B. Incorrect. First part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground. Second part is correct.

C. Correct. Per S2.OP-SO.125-0004, a resistance reading of 50K ohms or less indicates a ground is detected. A resistance of greater than 50K ohms is normal indicating no grounds are present. Therefore, a ground reading of 0 ohms indicates a hard ground is present. Per S2.OP-SO.125.0004, breakers on the bus will be opened one at a time until the resistance indicates greater than 50K ohms.

D. Incorrect. First part is correct. Second part is incorrect. Plausible because the operator can incorrectly recall and confuse which resistance reading indicates a ground present and believe that the 0 ohms indicates no grounds. Incorrect in that resistance reading less than 50K ohms is a ground.

Technical References:	NOS05DCELECT-11
Proposed References to be provided:	None
Learning Objective:	NOS05DCELEC-11, Objective 8.b. Identify and describe the Control Room controls, indications, and alarms associated with the DC Electrical System, including: The function of each DC Electrical System Control Room control and indication
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	06-DC Electrical Distribution System
K/A #:	063 K1.01 Knowledge of the physical connections and/or cause and effect relationships between the DC Electrical Distribution System and the following systems: Ground detection system.
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked how the ground detection system relates to the DC bus as far as the expected indications for a ground on the bus. Also, the candidate is asked how the actions in the procedure will determine the isolation of a ground.

47 **06-031** **Points: 1.00**

Which of the following completes both statements concerning the 125 VDC Electrical Distribution System?

1. The 125 VDC Distribution System design __(1)__ include a mechanical interlock when transferring from normal to the backup (emergency) bus.

2. Transferring 125 VDC distribution loads to the backup bus is permitted during __(2)__.

A. (1) does NOT
(2) ALL Modes

B. (1) does NOT
(2) Modes 5,6, or defueled Only

C. (1) does
(2) ALL Modes

D. (1) does
(2) Modes 5,6, or defueled Only

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator may incorrectly recall that the DC busses do not have a mechanical interlock. Incorrect in that a mechanical interlock is part of the design. Part 2 is correct. Part 2 is incorrect. Plausible because the operator may misinterpret the questions and believe that the DC bus can be transferred during any modes. Incorrect in that paralleling DC bus will make the DC busses Inoperable during Modes 1-4.

B. Incorrect. Part 1 is incorrect. Plausible because the operator may incorrectly recall that the DC busses do not have a mechanical interlock. Incorrect in that a mechanical interlock is part of the design. Part 2 is correct.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may misinterpret the questions and believe that the DC bus can be transferred during any modes. Incorrect in that paralleling DC bus will make the DC busses Inoperable during Modes 1-4.

D. Correct. The 125 VDC system does include a design feature that prevents paralleling the normal and backup busses. A mechanical interlock located in the distribution panel requires the normal supply breaker to be open and then the mechanical device is allowed to move allowing the closing of the backup or emergency power. Per procedure, transferring to the backup DC bus is only allowed during Modes 5, 6, or defueled.

Technical References:	NOS05DCELECT-11
Proposed References to be provided:	None
Learning Objective:	NOS05DCELECT-11, Objective 6.a. Outline the interlocks associated with the following DC Electrical System components: Normal and Emergency feeds to loads on125VDC buses
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	06-DC Electrical Distribution System
K/A #:	063 K4.01 Knowledge of DC Electrical Distribution System design features and/or interlocks that provide for the following: Manual/automatic transfers of control.
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked knowledge of 125VDC design feature with regards to a mechanical interlock used for transferring from normal to emergency buses.

48 **06-032** **Points: 1.00**

What are the power supplies to the 2A EDG starting air compressors 21A and 21B?

	21A	**21B**
A.	2A 230V Vital Bus	2A 230V Vital Bus
B.	2A 230V Vital Bus	2B 230V Vital Bus
C.	2A 460V Vital Bus	2A 460V Vital Bus
D.	2A 460V Vital Bus	2B 460V Vital Bus

Answer: A

Explanation / Justification:

A. Correct. Each EDG two starting air compressors are supplied from its associated bus 230 V vital bus. The 21A and 21B starting air compressors are both powered from 2A 230V Vital Bus.

B. Incorrect. Plausible because the operator may believe that the 21B starting air compressor is powered from the other EDG 230V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

C. Incorrect. Plausible because the operator may incorrectly recall and believe that the starting air compressors are powered from 2A 460V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

D. Incorrect. Plausible because the operator may believe that the 21B starting air compressor is powered from the other EDG 460V vital bus. Incorrect in that both starting air compressors are powered from the same 2A 230V vital power.

Technical References:	NOS05EDG000-15
Proposed References to be provided:	None
Learning Objective:	NOS05EDG000-15, Objective 5.e. State the power supply to the following Emergency Diesel Generator components: Starting Air Compressors
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	06-Emergency Diesel Generator System
K/A #:	064 K2.01 Knowledge of electrical power supplies to the following Emergency Diesel Generator System: Starting air compressor
Importance:	3.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked the power supplies to the EDG air compressors.

49 **07-026** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- The PO needs to perform a check source on the 1R19A, SGBD Radiation Monitor

Per S1.OP-ST.RM-0001, Radiation Monitors – Check Source, which of the following completes both statements?

1. When the operator pushes the check source pushbutton, the radiation monitor will respond by ___(1)__.
2. During the check source, actuation of interlocks __(2)__ expected.

A. (1) a rise in counts until the check source pushbutton is released
(2) are

B. (1) a rise in counts until the check source pushbutton is released
(2) are NOT

C. (1) a rise in counts then a return to original reading
(2) are NOT

D. (1) a rise in counts then a return to original reading
(2) are

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because other radiation monitors do perform like this. Incorrect in that this radiation monitor check source the counts rise then fall and interlocks are not expected during check source.

B. Incorrect. Plausible because other radiation monitors do perform like this. Incorrect in that this radiation monitor check source the counts rise then fall and interlocks are not expected during check source.

C. Correct. When a check source is performed on this radiation monitor, the counts will rise then return to its previous reading. During check source it is not expected for interlocks to actuate.

D. Incorrect. Plausible because some other radiation monitors interlocks could occur during check source.

Technical References:	NOS05RMS000-20, S1.OP-SO.RM-0001
Proposed References to be provided:	None
Learning Objective:	NOS05RMS000-20, Objective 3. Describe how the following components impact the Radiation Monitoring System during normal and abnormal conditions: R19A, B, C, & D, Steam Generator Blowdown Liquid Monitors
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	07-Process Radiation Monitor
K/A #:	073 A4.02 Ability to manually operate and/or monitor the Process Radiation Monitor in the control room: RMS control panel.
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking what the expected response at the radiation monitor panel for the 1R19 during check source operation.

50 **04-077** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power
- Three (3) Service Water (SW) Pumps are in service
- Four (4) CFCUs are running in HIGH speed
- SW header pressures are 108 psig

Then:

- The PO is directed to place one running CFCU in LOW speed for surveillance testing

Which of the following completes the statement concerning the response on the Service Water System (assume three SW pumps remain running at all times)?

As compared to the SW header pressure while operating in HIGH speed, the SW header pressure will be __(1)__ when operating in LOW speed and SW flow rate through the CFCU will be __(2)___ than before.

A. (1) lower
(2) the same

B. (1) lower
(2) higher

C. (1) the same
(2) the same

D. (1) the same
(2) higher

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the SW flow through the CFCUs is the same for high or low speed operation. Incorrect in that the SW header will be relatively the same.

B. Incorrect. Plausible because prior operation of the CFCUs in low speed use to have a higher flow rate. Incorrect in that orifices have been installed to maintain constant flow rates in either speed.

C. Correct. The SW flow rates through the CFCUs in High and Low speed are the same, therefore, the SW header pressure will also be the same when operating in High and Low speed.

D. Incorrect. Plausible because prior operation of the CFCUs in low speed use to have a higher flow rate. Incorrect in that orifices have been installed to maintain constant flow rates in either speed.

Technical References:	NOS05CONTMT-17
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 14. Given plant conditions, relate the Containment and Containment Support Systems with the following: Service Water
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	04-Service Water System
K/A #:	076 A1.03 Ability to predict and/or monitor changes in parameters associated with operation of the Service Water System: SWS header pressure
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the SW header pressure changes when the CFCU is swapped from High to Low speed.

51 **04-078** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 4
- 21 RHR Pump is in service providing shutdown cooling through 21 RHR HX
- RCS pressure is 300 psig
- 21RH18 and 2RH20 are being throttled to maintain a RCS cooldown rate of 5°F per hour

Then:

- Instrument air supply line to the actuator of 21SW122, SW Inlet Control Valve to CCW HX, severs

Assuming NO operator action, which of the following completes both statements?

1. RCS cooldown rate will ___(1)__.
2. The 21SW122 __(2)__ be manually operated.

A. (1) rise
(2) can NOT

B. (1) rise
(2) can

C. (1) lower
(2) can

D. (1) lower
(2) can NOT

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the operator may incorrectly recall if this valve can be manually operated from the field since not all AOVs can be manually operated in the field.

B. Correct. 21SW122 fails open on loss of air (fails closed on loss of 125VDC power). When the 21SW122 closes, SW flow will lower through the CCW HX causing CCW temperature to rise and therefore the outlet temperature of 21RHR HX to rise and lowering the cooldown rate. Per S2.OP-AB.CA-0001, the 21/22SW122 can be manually operated.

C. Incorrect. Plausible because the question is asking about the cooldown rate and the operator may confuse the concept that a lower temperature means a higher cooldown rate.

D. Incorrect. Plausible because the question is asking about the cooldown rate and the operator may confuse the concept that a lower temperature means a higher cooldown rate..

Technical References:	NOS05SW0NUC-13, S2.OP-AB.CA-0001 (R22)
Proposed References to be provided:	None
Learning Objective:	NOS05SW0NUC-13, Objective 4.f. Describe the function of the following components and how their normal and abnormal operation affects the Service Water – Nuclear Header System: CC HX SW122
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.6
Level:	RO
Category/System:	04-Service Water System
K/A #:	076 K3.05 Knowledge of the effect that a loss or malfunction of the Service Water System will have on the following systems or system parameters: RHRS
Importance:	3.6
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant how the SW122 service water control valve for the CCW HXs will impact the RHR system and the RCS cooldown rate..

52 **08-022** **Points: 1.00**

Given:

- Salem Unit 2 experienced a Safety Injection
- The crew enters 2-EOP-TRIP-1, Reactor Trip or Safety Injection
- Phase A signal failed to actuate on BOTH trains

Which of the following completes the statement concerning instrument air to containment?

1. 21 and 22 CA330s, Instrument Air Containment Isolation, valves are currently in the __(1)__ position.
2. MINIMUM requirement per 2-EOP-TRIP-1 for isolating instrument air to containment requires __(2)__ CA330(s) to be closed.

A. (1) closed
(2) both

B. (1) closed
(2) Only one

C. (1) open
(2) Only one

D. (1) open
(2) both

Answer: D

Explanation / Justification:

A. Incorrect Part 1 is incorrect. Plausible because the operator may believe that the CA330s close on an SI signal. Incorrect in that each CA330s will close on a Phase A signal only. Part 2 is correct. Plausible because the operator may believe that the CA330s are in series and need both to close

B. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that the CA330s close on an SI signal. Incorrect in that each CA330s will close on a Phase A signal only. Part 2 is incorrect. Plausible because the operator may believe that the CA330s are in series and only need one of the two to isolate containment as a minimum.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may believe that the CA330s are in series and only need one of the two to isolate containment as a minimum.

D. Correct. CA330s are closed by a Phase A isolation signal. Although SI actuated, with both trains of Phase A not actuated, then the 21 and 22 CA330s will remain open. Both 21 and 22 CA330s are required to be closed to fully isolate air to containment. 21CA330s supplies 2A control air header and 22CA330s supplies 2B control air header to containment. Each instrument air supply line to containment is isolated by its respective CA330 and a check valve. It is for this reason that both CA330s need to be closed to ensure a failure of one of the two components will still isolate a pathway from containment. Backup air receivers inside containment ensure that the PZR PORVs have sufficient air to allow the PZR PORV to stroke during steps in the EOP Network (Inadvertent SI, SGTR, etc.)

Technical References:	NOS05CONAIR-13
Proposed References to be provided:	None
Learning Objective:	NOS05CONAIR-13, Objective 4. Describe how each impact the Control Air System during normal and abnormal conditions: CA Containment Isolation Valves
Question Source:	Modified Bank – Farley 2019 NRC RO14
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	08-Instrument Air System
K/A #:	078 A3.05 Ability to monitor automatic features of the Instrument Air System, including: Isolation of instrument air to containment
Importance:	3.3
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant is being asked how to verify the automatic feature (Phase A signal) closed the CA330s. In this case, with no Phase A signal present , then both CA330s will remain open.

Parent Question: Farley 2019 NRC RO14

Modified Part 2 of the question.

Question # 14

Given the following:

- Unit 3 experiences an SI from 100% power.
- 3-EOP-E-0, Reactor Trip or Safety Injection, is in progress.

Which one of the following completes the statements below?

CV-3-2803, Instrument Air Containment Isolation is expected to be in the __(1)__ position.

Instrument Air supply to containment is essential for operation of __(2)__ in subsequent steps in the EOP Network.

A. (1) OPEN
(2) Pressurizer Spray valves.

B. (1) OPEN
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

C. (1) CLOSED
(2) Pressurizer Spray valves.

D. (1) CLOSED
(2) HCV-3-121, Charging Flow To Regen Heat Exchanger.

Answer Analysis

Containment Isolation Signal does not close the Instrument Air to Containment Isolation valve at PTN. As per the Design Basis. Also, the design basis states that the verification of IA to Containment is prioritized in the procedure to ensure equipment in subsequent steps can be operated. Pressurizer spray is needed to stabilize RCS pressure when SI Termination criteria is met.

A. (1) Correct
(2) Correct

B. (1) Correct
(2) Incorrect, plausible because the HCV-121 is required to be operated when restoring Charging flow per EOP-2 Attachment 2, however this valve gets Instrument Air from the Aux Building IA Header.

C. (1) Incorrect, Plausible since this is a containment isolation valve, however, it does not get a CIV signal and fails open.
(2) Correct

D. (1) Incorrect, see C(1)
(2) Incorrect. See (B(2)

53 **05-025** **Points: 1.00**

Which of the following automatically generated signals will isolate the Containment Vacuum Relief Unit from the containment atmosphere when in service?

1. Phase A
2. Safety Injection
3. Containment Vent Isolation
4. Containment Spray/Phase B

A. 1 and 2 Only

B. 2 and 3 Only

C. 2, 3, and 4 Only

D. 1, 3, and 4 Only

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because Phase A signals do close penetrations through containment. Incorrect in that the Phase A signal does not actuate a CVI signal.

B. Correct. The Containment Vacuum Relief System filter unit uses only roughing filters. Since air is being drawn into containment there is no need to filter the air for radioactive contaminants. The dampers (VC5 and VC6) will automatically close on an automatic Safety Injection or CVI signal. The CVI signal will be actuated from a Safety Injection. CVI signal can also be generated manually using either (1) CS/Phase B keyswitches, or (2) Containment Vent Operate pushbutton on the safeguards console. Containment Atmosphere Radiation Monitors 2R11A, 2R12A, and 2R12B either one in alarm will cause a CVI.

C. Incorrect. Plausible because manually actuating CS/Phase B will actuate a CVI signal. Incorrect in that the stem is asking about automatically generated signals only.

D. Incorrect. Plausible because Phase A and B do isolate containment penetrations. Incorrect in that these signals does not actuate a CVI.

Technical References:	NOS05CONTMT-17, 221057 (RPS logic)
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 5. Outline the interlocks associated with the Containment Pressure/Vacuum Relief System containment isolation valves, VC-5 and VC-6
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7
Level:	RO
Category/System:	05-Containment System
K/A #:	103 K4.01 Knowledge of Containment System design features and/or interlocks that provide for the following: Vacuum breaker protection
Importance:	3.2
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the applicant knowledge of the design features of the containment vacuum relief system used to relief negative pressure in containment.

Additional Reference: NOS05CONTMT

c. Indications

1) The Pressure Relief Unit has local indication of:

a) Roughing Filter differential pressure (PL-7690)

b) HEPA Filter differential pressure (PL-7689)

c) Charcoal Filter differential pressure (PL-7688)

2) The Vacuum Relief Unit has local indication of Roughing Filter differential pressure (PL-7691)

3) Local position indication is provided for:

a) CP - VRS inside containment isolation valve VC-6

b) CP - VRS outside containment isolation valve VC-5

c) Pressure Relief Unit Outlet Damper CBV36

d) Vacuum Relief Unit Outlet Damper CBV35

Additional Reference:

ELO 8.b

3) Containment Ventilation Isolation (CVI)

a) Initiated by:

- Safety Injection actuation
- High Containment atmosphere particulate, Iodine, or gaseous activity or High Plant Vent Release Rate
  - R11A, R12A & B, and R41D High Radiation alarm will initiate a CVI
- Manual actuation from the Control Room of:
  - Containment Ventilation Actuation
  - Phase B Containment Isolation/Containment Spray Actuation

b) The CVI signal will close:

- VC1, VC4, VC5, & VC6
- Note: VC2 & VC3 have been replaced by blind flanges VCF2 & VCF3. These flanges are the containment boundary in modes 1 through 4. They are removed to support containment purge in modes 5,6, & defueled.

c) The CVI function is required in Modes 1-6

54 **05-026** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A steam leak is occurring inside containment
- Containment pressure is slowly rising

Based on the above condition, which of the following completes both statements?

1. If containment pressure exceeds (+) __(1)__ psig, then containment DESIGN LIMITS could be exceeded during a Design Based Accident.

2. TS 3.6.1.4, Internal Pressure, requires containment internal pressure to be restored within the limits within __(2)__.

A. (1) 1.5
(2) 1 hour

B. (1) 1.5
(2) 15 minutes

C. (1) 0.3
(2) 15 minutes

D. (1) 0.3
(2) 1 hour

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse this value with the TS value of (-) 1.5 psig.

B. Incorrect. Plausible because the operator may confuse this value with the TS value of (-) 1.5 psig.

C. Incorrect. Plausible because there are other TS LCO with less than one hour limits.

D. Correct. Based on TS bases for TS 3.6.1.4, Internal Pressure must be maintained between -1.5 and +0.3 psig during Modes 1-4. Limitation on containment internal pressure ensure that (1) the containment structure Is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break condition. The pipe breaks considered are LOCA and steam line break.

Technical References:	TS 3.6.1.4 and Bases
Proposed References to be provided:	None
Learning Objective:	NOS05TECHSPEC-14, Objective 12. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	05-Containment System
K/A #:	103 K6.15 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Containment System: MRSS
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the applicant must have knowledge of how a steam leak from the Main Steam system will affect the TS assumption and limits of containment internal pressure.

55 **01-018** **Points: 1.00**

Given:

- Salem Unit 2 is performing a load reduction from 100% power
- Rod Control is in AUTO

Then:

- Console Alarm, URGENT FAILURE, is Alarming
- Rod motion has stopped

Which of the following completes both statements concerning the effect on the rod control system?

1. Rods __(1)__ be inserted in MANUAL.
2. The rods are currently held in position by energizing the __(2)__ gripper coils.

A. (1) can
(2) stationary

B. (1) can
(2) moveable

C. (1) can NOT
(2) moveable

D. (1) can NOT
(2) stationary

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator could believe that even with an Urgent Failure present, the operator will always be able to insert rods. Incorrect in that all rod motion is inhibited in Manual and Automatic. Part 2 is correct.

B. Incorrect. Part 1 is incorrect. Plausible because the operator could believe that even with an Urgent Failure present, the operator will always be able to insert rods. Incorrect in that all rod motion is inhibited in Manual and Automatic. Part 2 is incorrect. Plausible because the moveable gripper coil is used to hold the rods in position until the lift coil is energized/de-energized to withdraw or insert rods. Incorrect in that the stationary coil is energized at a reduced current to hold rods in position.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the moveable gripper coil is used to hold the rods in position until the lift coil is energized/de-energized to withdraw or insert rods. Incorrect in that the stationary coil is energized at a reduced current to hold rods in position.

D. Correct. Per console ARP S2.OP-AR.ZZ-0012, Urgent Failure alarm will prevent all rod motion in Manual or Automatic. In this condition the rods are held in position by the stationary gripper coil being energized at a reduced current. The moveable gripper coil is de-energized at this time.

Technical References:	NOS05RODS00-14, S2.OP-AR.ZZ-0012 (R40)
Proposed References to be provided:	N/A
Learning Objective:	NOS05RODS00-14, Objective 6.b. Describe the function of the following components and how their normal and abnormal operations affects the Rod Control and Position Indication Systems: Control Rod Drive Mechanism (CRDM)
Question Source:	Modified Bank – Diablo Canyon 2019 NRC RO29
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.2 to 41.9 / 45.7 to 45.8
Level:	RO
Category/System:	01-Control Rod Drive System
K/A #:	001 K1.03 Knowledge of the physical connections and/or cause and effect relationships between the Control Rod Drive System and the following systems: CRDM
Importance:	3.7
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the operator knowledge of how the CRDM is effected when an Urgent Failure alarm is present and which gripper coil is used to hold the rods in position.

Modified Question: Part 2 of the parent question was modified to ask which gripper coli is energized to hold the rods in-place.

Parent Question: Diablo Canyon 2019 NRC RO29

Question 29

Unit 2 is ramping power down from 100% power.

PK03-17, Rod Cont Urgent Failure, alarms.

1) Rods _____ be inserted in MANUAL.
2) The lift coil will be _____.

A. 1) can
   2) energized at reduced current

B. 1) can
   2) deenergized

C. 1) can NOT
   2) energized at reduced current

D. 1) can NOT
   2) deenergized

Proposed Answer: D. 1) can NOT 2) deenergized

Additional Reference: NOS05RODS00 Rod Control Lesson Plan

e) Coil Current Levels

(1) 3 Levels of current supplied to coils

(2) HIGH–Applied when coil is initially energized to ensure positive latching, then reduced

- Lift coil–40 amps
- Movable/stationary. Gripper coil–8 amps

(3) LOW–Applied during remainder of stepping sequence. Also applied to stationary coils to hold rods

- Lift coil–16 amps
- Movable gripper coil is not reduced
- Station Gripper Coil–4.4 amps

(4) OFF–coil deenergized

56 **01-019** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Control Bank D rods are at 200 steps

Then:

- The Group Demand Step Counter on 2CC2 console for Control Bank D Group 1 rods fails (blank screen)

Which of the following completes both statements?

1. The Plant Computer displays the rod demand step position for Group(s) __(1)__.
2. Based on the above condition, the Plant Computer __(2)__ be used to satisfy the Tech Spec requirements for the failed console group demand step counter.

A. (1) 1 Only
(2) can

B. (1) 1 and 2
(2) can NOT

C. (1) 1 and 2
(2) can

D. (1) 1 Only
(2) can NOT

Answer: A

Explanation / Justification:

A. Correct. . Only the Group 1 Demand Steps input into the Plant Computer. Per S2.OP-DL.ZZ-0003 for Control Rod Counter channel checks, either the Group Demand Counter or Plant Computer will satisfy the surveillance for Group 1 only. Only the Group 1 Demand Steps input into the Plant Computer. TS 3.1.3.2.1.b, bases states in part, "…either the control console group 1 demand step counter or the plant computer "bank step" display is sufficient to comply with this specification for group 1 rod position."

B. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that both groups are also displayed on the plant computer. Part 2 is incorrect. Plausible because the operator may believe that the Plant Computer cannot satisfy the surveillance requirement. Incorrect in that per S2.OP-DL.ZZ-0003, the Plant Computer Group Demand for Group 1 only will satisfy the surveillance and also that only group 1 demand is displayed on the plant computer.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that both groups are also displayed on the plant computer. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may believe that the failed step counter will generate E-24. Incorrect in that E-24 does not use input directly from the console step counters. Part 2 is incorrect. Plausible because the operator may believe that the Plant Computer cannot satisfy the surveillance requirement. Incorrect in that per S2.OP-DL.ZZ-0003, the Plant Computer Group Demand for Group 1 only will satisfy the surveillance.

Technical References:	S2.OP-DL.ZZ-0003 (R126), TS 3.1.3.2.1 bases
Proposed References to be provided:	None
Learning Objective:	NOS05RODS00-14, Objective 6, Describe the function of the following components and how their normal and abnormal operations affects the Rod Control and Position Indication Systems: Rod Insertion Limit Comparator
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	RO
Category/System:	01-Position Indication System
K/A #:	014 A2.11 Ability to (a) predict the impacts of the following on the Position Indication System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failed group demand position indication
Importance:	3.0
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the candidate is being asked to predict how the console step counter will effect rod position system and determine if the channel check surveillance can be satisfied using the plant computer.

Additional References: S2.OP-DL.ZZ-0003 Control Room Logs Modes 1-4

ATTACHMENT 1 S2.OP-DL.ZZ-0003(Q)

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MONDAY DATE: ___/___/___

CONTROL ROOM LOG (MODES 1-4)

DESCRIPTION	SURV. RQMT	INSTR NO.	UNITS	MIN	MAX	MON 1830	MON 0630	TUE 1830	TUE 0630	WED 1830	WED 0630	THU 1830	THU 0630	FRI 1830	FRI 0630	SAT 1830	SAT 0630	SUN 1830	SUN 0630
SHUTDOWN ROD COUNTERS																			
ALL ROD DEMAND COUNTERS/IRPI STEPS PER PLANT COMP	4.1.3.4.b	---	STEPS	UNSAT	SAT														
* IF ALL Plant Computer IRPI indicate ±12 steps (for >85% RTP) OR ±18 steps (for <=85% RTP) of their respective group demand counters, THEN RECORD "SAT". IRPI indication of <0 steps or >235 steps are considered 0 or 235 steps respectively.																			
CONTROL ROD COUNTERS																			
GROUP NOTE - S/R 4.1.3.1.1, 4.1.3.2.1.1 & 4.1.3.5 - Maintain Control Bank Overlap of 101 Steps. IF ALL rods in the indicated bank are within ±12 steps (for >85% RTP) OR ±18 steps (for <=85% RTP) of the group step counters, THEN RECORD "SAT". IF Reactor Trip Breakers are open, THEN RECORD "STBY". IF in Mode 1 or 2 AND EITHER the Rod Position Deviation Alarm (OHA-E24) OR the Rod Insertion Limit Monitor is inoperable, THEN RECORD Step Counter and Stepcounter/IRPI comparison every 4 hours on Attachment 4. GROUP NOTE - FOR Group Demand Counter: Either the Group Counter or the Plant Computer will satisfy the surveillance for Grp 1 only. FOR IRPI INDICATION: Either the Dixon or the Plant Computer will satisfy the surveillance. Both are recorded to provide flexibility if an indication fails. IRPI indication of <0 steps or >235 steps are considered 0 or 235 steps respectively. IF an IRPI (Dixon or Plant Computer) is UNSAT, THEN RECORD in the comments, the specific rod(s) which are affected, THEN GO TO S2.OP-AB.ROD-0001(Q).																		
BANK A DEMAND	4.1.1.1.b	---	STEPS	UNSAT	SAT														

57 **07-027** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The ventilation system that supplies cool air to the room that the Reference Junction Panel is located for the In-Core Temperature Monitoring System is lost
- Room temperature have risen by 15 °F

Which of the following describes the impact on the In-Core Monitoring instrumentation?

A. CET readings will indicate higher than actual and can be used for trending purposes up to 3300 °F.

B. CET readings will indicate higher than actual and can be used for trending purposes up to 2300 °F.

C. NO effect on CET readings as they are temperature compensated and can be used for trending purposes up to 2300 °F.

D. NO effect on CET readings as they are temperature compensated and can be used for trending purposes up to 3300 °F.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because higher temperatures in the room where the reference junction box is located will cause the in temperature indicated to rise. Incorrect in that the system has a temperature compensator circuit such that the reference junction box temperature changes are adjusted to the true thermocouple reading. In-core system has a design range to 2300 °F.

B. Incorrect. . Plausible because higher temperatures in the room where the reference junction box is located will cause the in temperature indicated to rise. Incorrect in that the system has a temperature compensator circuit such that the reference junction box temperature changes are adjusted to the true thermocouple reading. In-core system has a design range to 2300 °F.

C. Correct. The reference junction box is located outside of containment. Ambient temperature changes in the room will affect the reference junction box, however the system uses a temperature compensator to adjust for the changes in temperatures where the reference junction box is located. A rise in ambient room temperature will raise temperature, but the temperature compensating circuit will adjust for these reference junction box temperature changes.

D. Incorrect. In-Core system design upper range is 2300 °F.

Technical References:	INPO Sensors and Detectors (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05INCORE-09, Objective 7. Describe how the following components affect the Core Exit Thermocouple System during Normal and Abnormal conditions: Reference Junction Boxes
Question Source:	Bank – Indian Point 2 2021 NRC RO58
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	07-In Core Temperature Monitoring System
K/A #:	017 K6.01 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the In Core Temperature Monitoring System: Temperature measuring device (for example thermocouple)
Importance:	3.2
Tier/Group	Emergency and Abnormal Plant Evolutions -Tier 2/Group 2

K/A Match: The K/A is matched because the candidate is being asked how the In Core Monitoring System is impacted based on ambient temperatures changes at the reference junction box.

Additional References: INPO Sensors and Detectors

Thermocouple Operation Example

Thermocouples will cause an electric current to flow in the attached circuit when subjected to changes in temperature. The amount of current produced depends on the temperature difference between the measurement and reference junction, the characteristics of the two metals used, and the characteristics of the attached circuit. The figure below shows a basic thermocouple circuit.

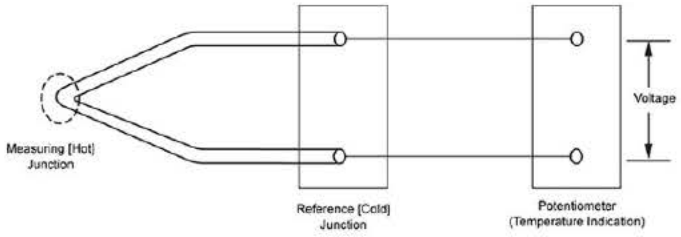


Figure: Simple Thermocouple Circuit

Heating the measuring junction of the thermocouple produces a voltage that is greater than the voltage across the reference junction. The voltmeter measures the difference between the two voltages (in millivolts); the voltage is proportional to the difference in temperature.

$$Temperature\ Indicated \propto Hot\ Junction\ Voltage - Cold\ Junction\ Voltage$$

If temperature at the reference (cold) junction were to decrease, indicated temperature would increase and vice versa. For ease of operator use, some voltmeters are set up to read out directly in temperature through use of electronic circuitry. Other applications provide only the millivolt readout. In order to convert the millivolt reading to its corresponding temperature, the operator must refer to vendor-supplied thermocouple tables. The thermocouple manufacturer supplies these tables, and they list the specific temperature corresponding to a series of millivolt readings.

Additional Reference: NOS05INCORE-06

LESSON NAME: INCORE NUCLEAR INSTRUMENTATION SYSTEM, INCLUDING THERMOCOUPLES AND SATURATION MARGIN MONITOR
NOS05INCORE-06 04/09/19

INSTRUCTIONAL CONTENT:

2) To measure the cold junction temperature with a semiconductor temperature probe, and provide that temperature signal to the CETPS so that the temperature error introduced by the cold junction may be compensated.

NOTE: The cold junction temperature is the temperature at the point where the thermocouple wires are connected to copper wires at the terminal strip. This is different than the temperature sensed by the thermocouple, because, in essence, a new thermocouple is formed by the junction between the thermocouple wires and the copper wire. This "new" thermocouple produces a voltage proportional to its junction temperature, which adds to the voltage already present from the FTTC. By measuring this temperature, the CETPS can compensate for this error to produce a true core temperature.

58 **05-027** **Points: 1.00**

Which of the following completes both statements concerning operation of the Hydrogen Recombiner System during a Design Bases Accident (assume hydrogen concentration is within the range that allows operation of the hydrogen recombiner system)?

1. The maximum number of hydrogen recombiners that can be placed in service is __(1)__.
2. What is the concern if hydrogen recombiner system cannot be placed in service when required?

A. (1) 1
(2) Gas vapor bounding of ECCS pumps resulting in loss of core cooling.

B. (1) 1
(2) Explosive atmosphere resulting in containment failure.

C. (1) 2
(2) Explosive atmosphere resulting in containment failure.

D. (1) 2
(2) Gas vapor bounding of ECCS pumps resulting in loss of core cooling.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because gas intrusion into the ECCS system can adversely affect the operation of ECCS pumps.

B. Correct. Per 2-EOP-LOCA-1 step 17, only one Hydrogen Recombiner is allowed to be placed in service when the Hydrogen concentration is between > 0.5 % and < 4%. The bases for running a Hydrogen Recombiner is to reduce the hydrogen concentration to prevent the formation of a flammable mixture that could ignite and cause an explosion and pressure excursion that could challenge containment integrity.

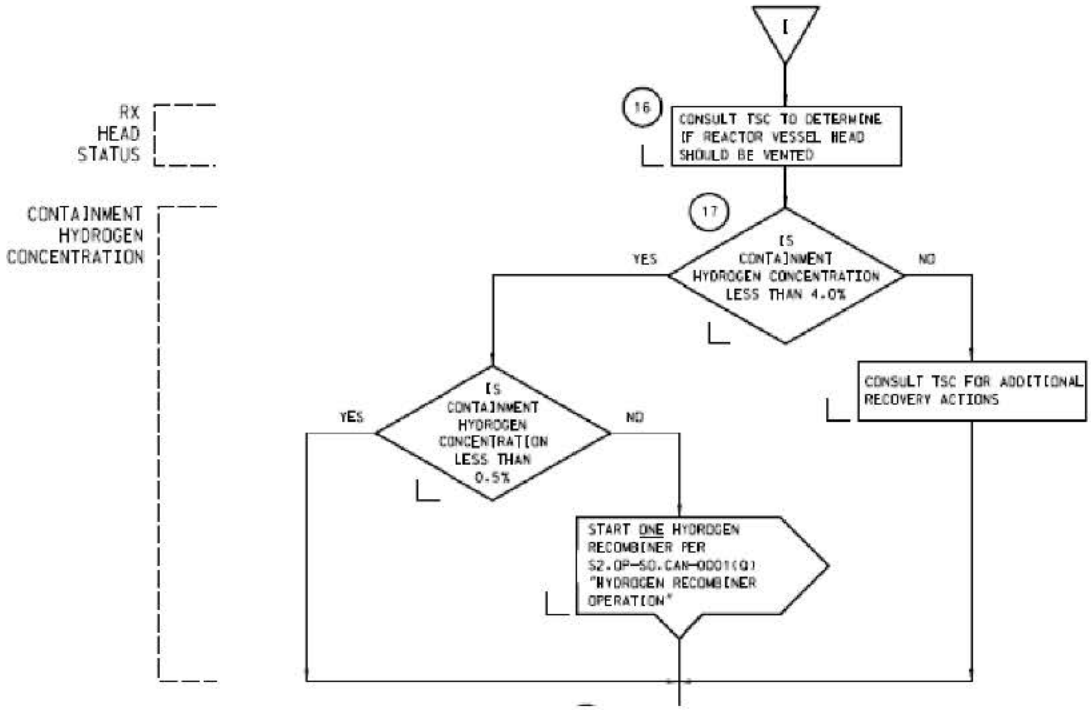
C. Incorrect. Plausible because the operator may believe that both hydrogen combiners can be operated to maximize the reduction of hydrogen when hydrogen concentration is at the upper end (4%). Incorrect in that operation of both hydrogen recombiners is not permitted.

D. Incorrect. Plausible because gas intrusion into the ECCS system can adversely affect the operation of ECCS pumps. Plausible because the operator may believe that both hydrogen combiners can be operated to maximize the reduction of hydrogen when hydrogen concentration is at the upper end (4%). Incorrect in that operation of both hydrogen recombiners is not permitted.

Technical References:	2-EOP-LOCA-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA01-07, Objective 6.j. Describe the plant response to actions taken in the following EOP step sequence(s): Containment Hydrogen Concentration
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.4
Level:	RO
Category/System:	05-Hydrogen Recombiner and Purge Control System
K/A #:	028 K3.02 Knowledge of the effect that a loss or malfunction of the Hydrogen Recombiner and Purge Control System will have on the following systems or system parameters: CNT
Importance:	2.9
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant what the effect on containment will be when hydrogen recombiners cannot be placed in service when hydrogen concentration requires it in EOP-LOCA-1.

Additional Reference: 2-EOP-LOCA-1



Additional Reference:

ERG Step No: Step 17

EOP Step:

IS CONTAINMENT HYDROGEN CONCENTRATION LESS THAN 4.0%
[CONTAINMENT HYDROGEN CONCENTRATION]

Purpose:

To check if an excessive containment hydrogen concentration is present.

ERG Basis:

This step instructs the operator to obtain a current hydrogen concentration measurement. Depending upon the magnitude of the hydrogen concentration, the operator will either continue with EOP-LOCA-1, turn on the hydrogen recombiners or notify the plant engineering staff to determine additional recovery actions before continuing with the procedure.

When inadequate core cooling has occurred, the containment hydrogen concentration may be as much as 10 to 12 volume percent, depending on the amount of metal-water reaction (to produce hydrogen) that has occurred in the core. The hydrogen concentration is of concern since a flammable mixture can burn, if an ignition source is available, and cause a sudden rise in containment pressure which may challenge containment integrity. The operator is instructed to obtain a current containment hydrogen concentration measurement at this point in order to ascertain the potential flammability of the combustible gases in the containment. Note that in order to have the potential for flammable hydrogen concentrations, an inadequate core cooling situation must have already existed. Without an inadequate core cooling situation, sufficient hydrogen would not be expected to have been produced to cause potentially flammable mixtures.

A determination is made of the flammability of the hydrogen mixture with respect to the possible containment pressure rise. If the hydrogen mixture is between 0.5 volume percent and 6.0 volume percent in dry air, either no hydrogen burn is possible or a limited burn may occur which does not produce a significant pressure rise. A hydrogen concentration not to exceed 6.0 volume percent in dry air corresponds to the upper limit of operability for the hydrogen recombiner, represented by the footnote (T.18). If containment hydrogen concentration is between 0.5 volume percent and(T.18) volume percent, the operator is instructed to start the Hydrogen Recombiner System to slowly reduce containment hydrogen concentration. If the hydrogen concentration is less than 0.5 volume percent in dry air, a flammable situation is not imminent and the operator continues with procedure EOP-LOCA-1. If the concentration is greater than (T.18) volume percent in dry air, the operator is instructed to immediately notify the plant engineering staff of the situation. In this case the operator is instructed to consult the plant engineering staff for additional recovery actions while proceeding with this procedure.

59 **08-023** **Points: 1.00**

Given:

- Salem Unit 1 is in MODE 5
- Spent Fuel Pool level is 128' 5"
- Spent Fuel Pool boron concentration is 2300 ppm

Then:

- A tube inside the in-service Spent Fuel Pool Heat Exchanger ruptures

Which of the following completes both statements?

1. Level in the Spent Fuel Pool will __(1)__.
2. The Spent Fuel Pool boron concentration will require adjustments from a __(2)__ water source.

A. (1) rise
(2) non-borated

B. (1) rise
(2) borated

C. (1) lower
(2) borated

D. (1) lower
(2) non-borated

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SFP level will rise. Incorrect in that CCW is a non-borated water source and will reduce the boron concentration.

B. Correct. Per S2.OP-AB.CC-0001, a leak in the SFP HX will result in the out leakage of CCW into the SFP system resulting in the CCW Surge Tank level lowering and the SFP level rising. The non-borated CCW water will reduce the boron concentration of the SFP. If boron concentration lowers below TS then the crew will need to add borated water into the SFP to raise the boron concentration. Note: SFP low and high level alarms 128' 2" and 129' 2".

C. Incorrect. Plausible because the operator may incorrectly recall that the SFP HX will leak into the CCW system. Incorrect per S1.OP-AB.CC-0001, SFP HX will be out leakage from CCW.

D. Incorrect. Plausible because the operator may incorrectly recall that the SFP HX will leak into the CCW system. Incorrect per S1.OP-AB.CC-0001, SFP HX will be out leakage from CCW and that the boron concentration will reduce requiring addition from a borated source..

Technical References:	S1.OP-IO.ZZ-0010 (R30), TS 3.9.11
Proposed References to be provided:	None
Learning Objective:	NOS05SFP000-12, Objective 5.e. Describe the function of the following components and how their abnormal operation affects the Spent Fuel Pool Cooling System: Spent Fuel Pool Heat Exchanger
Question Source:	Bank – Turkey Point 2016 NRC RO61
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.7
Level:	RO
Category/System:	08-Spent Fuel Pool Cooling System
K/A #:	033 K6.11 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Spent Fuel Pool Cooling System: CCWS
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant what the effect will be on the SFP when a tube leak exists in the SFP heat exchanger. In this case, CCW will leak into the SFP system.

Additional References: Turkey Point 2016 RO61

Question # 61

Given the following initial conditions:

- Unit 3 is in Mode 5.
- Spent Fuel Pit (SFP) level is 56' 11".
- SFP boron concentration is 2300 ppm.

Subsequently:

- A tube inside of the in-service Spent Fuel Pool Heat Exchanger breaks.
- ANN H 8/6 CCW HEAD TANK HI/LO LEVEL alarms.
- A crew is dispatched to isolate the heat exchanger.

Which one of the following correctly completes the statement below?

The crew must take action to prevent Spent Fuel Pool ___(1)___ from violating the Tech Spec LCO by performing a ___(2)___ to the SFP in accordance with 3-NOP-033, Spent Fuel Pit Cooling System.

A. (1) level
(2) direct boration

B. (1) level
(2) primary water fill

C. (1) boron concentration
(2) direct boration

D. (1) boron concentration
(2) primary water fill

Additional Reference: S1.OP-IO.ZZ-0010

ATTACHMENT 1
(Page 5 of 5)
SUPPORT SYSTEMS STATUS

3.0 SYSTEMS/COMPONENTS REQUIRED OPERABLE:

System/Component (14)		Actual Status	Notes	Comments
FHB VENTILATION	(10)			
FUEL HANDLING CRANES			TRM 3.9.7	
SFP BORON CONCENTRATION	(11)	Cb: ____ Date: ____	Cb Required: T/S 3.7.11, T/S 3.9.1, FSAR 9.1.3	
CONTROL ROOM EMERGENCY AIR CONDITIONING SYSTEM	(12)		T/S 3.7.6.1	
1R1B RAD MONITORING	(13)		T/S 3.3.3.1	
CHILLED WATER SYSTEM			T/S 3.7.10	

(10) The FHB Supply Fan AND both FHB Exhaust Fans should be AVAILABLE AND operating. Each FHB Exhaust Fan is to be aligned to an available Vital Bus. A FHB Exhaust Fan may be aligned to 1C Vital Bus as an alternate power source IAW S1.MD-FR.ZZ-0001(Q), Alternate Power Sources During Refueling Outages. IF Fuel Handling Ventilation becomes unavailable, operations may continue after the Fuel Handling Area ventilation system has been evaluated and building D/P verified to be negative.

(11) The SFP Boron concentration (Cb) shall be greater than or equal to800 ppm. Applicable when fuel assemblies are stored in the SFP and a fuel storage pool verification has NOT been performed since the last movement of fuel assemblies in the SFP (T/S 3.7.11).
In Mode 6, the SFP Boron concentration (Cb) shall be within the limits specified in the Core Operating Limits Report (COLR). Applicable when the SFP is connected to the Reactor Coolant System (T/S 3.9.1).
For Dry Cask Storage Campaigns, SFP required Boron concentration (Cb) shall be provided by Reactor Engineering IAW SC.RE-FR.DCS-0002(Q).
The SFP Boron concentration (Cb) is maintained at a minimum of 800 ppm (FSAR 9.1.3).

(12) For fuel movement, the CREACS fans do NOT require an Emergency Backup power supply to be considered OPERABLE.

(13) Radiation monitors are inoperable with the system in ACCIDENT PRESSURIZED mode. This is an acceptable condition.

(14) **REFER** to S1.OP-SO.DG-0005(Q), Preparation For Removing A Diesel Generator From Service, for "Required Minimum Components Necessary For Movement Of Irradiated Fuel", based on Emergency Diesel Generator status.

Additional Reference: S1.OP-AB.CC-0001

___ 3.35 **CONTINUE** efforts to locate and isolate the leak, as applicable.

___ 3.36 **EVALUATE** initiating Unit shutdown IAW the applicable IOP.

NOTE

___ Components supplied by CCW with pressure less than CCW HDR pressure are sources for system to system out leakage (e.g., Spent Fuel Pool Heat Exchanger).

___ 3.37 **ISOLATE** the affected Component using ATTACHMENT 3, Equipment Isolation Table.

3.38 **WHEN** source of component leakage has been identified and isolated,

___ **THEN ENSURE** any valves that were CLOSED to support leak detection IAW ATTACHMENT 4, Leak Detection Method, are realigned to support normal Component Cooling System operation.

___ 3.39 **NOTIFY** SM/CRS to refer to the following:

___ ♦ Technical Specifications

___ ♦ Event Classification Guide

60 **04-079** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- A tube in one of the Circulating Water (CW) waterbox completely severs

Which of the following control room console alarms would confirm a gross tube leak in a CW waterbox?

1) Dissolved Oxygen Hi
2) Hydrazine Lo
3) Condensate Pump Disch Sodium Hi
4) Hotwell Outlet Conductivity Hi

A. 1 and 3 Only

B. 2 and 4 Only

C. 1, 3, and 4 Only

D. 2, 3, and 4 Only

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because these two alarms would confirm a waterbox tube leak. Incorrect in that the answer is incomplete and missing Hotwell Outlet Conductivity Hi.

B. Incorrect. Plausible because Dissolve Oxygen alarm would be expected and Hydrazine Lo alarm is an alarm for the feedwater systen. Incorrect in that it's not expected for river water into the condensate system to impact the concentration of Hydrazine.

C. Correct. Based on a severed tube in a waterbox, these alarms would confirm a tube leak in a CW waterbox.

D. Incorrect. Plausible because these alarms would confirm a tube leak. Incorrect in that it's not expected for river water into the condensate system to impact the concentration of Hydrazine.

Technical References:	S2.OP-AB.CHEM-0001 (R30)
Proposed References to be provided:	None
Learning Objective:	NOS05CONTMT-17, Objective 3.g,
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.3 / 45.12
Level:	RO
Category/System:	04-Condensate System / Emergency Procedures/Plan
K/A #:	056 G2.4.46 Ability to verify that the alarms are consistent with the plant conditions
Importance:	4.2
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant to verify which alarms in the control room would be present on the Condensate System if a CW waterbox tube leak exists.

Additional Reference:

S2.OP-AB.CHEM-0001(Q)

ATTACHMENT 2
CONDENSER TUBE LEAK
(Page 1 of 5)

1.0 **Actions for Condenser Tube Leak**

NOTE

___ A gross Condenser Tube Failure/Leak is indicated by rapidly rising Secondary Plant Chemistry recorder trends, 2CC1 and OHA alarms, it is also defined as:

___ ≥30 ppb Condensate Pump Disch sodium with condensate polisher in service, OR

___ ≥2 ppb Condensate Pump Disch sodium with condensate polisher out of service, OR

___ A rapidly increasing Condensate Pump discharge sodium in either condition (condensate polishers in or out of service) that is expected to exceed the limits above (>30 or >2 ppb).

2RP5 Recorders

___ Conductivity 2KA-8551
___ Sodium, Specific CNDCT

2CC1 Alarms

___ CONDENSATE PUMP DISCH SODIUM HI
___ HOTWELL OUTLET CONDUCTIVITY HI
___ CONDENSATE PUMP DISCH CONDUCTIVITY HI

OHA Alarms

___ COND POL AUTO SWAP (G-19)

61 **09-007** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power and Unit 2 is in MODE 5.
- Release of 11 CVCS Monitor Tank is in progress.
- The release path is through 11 CC Heat Exchanger and Unit 2 CW system
- 22A and 22B Circulators are the only circulators in service

Then:

- 22A Circulator trips

Based on the above condition, which one of the following describes the impact to the liquid release in progress?

1) The release __(1)__ automatically isolate upon the loss of the circulator.
2) The release __(2)__.

A. (1) will NOT
(2) may continue

B. (1) will NOT
(2) may NOT continue

C. (1) will
(2) may NOT recommence

D. (1) will
(2) may recommence

Answer: A

Explanation / Justification:

A. Correct. Per S1.OP-SO.WI-0001 only one circulator is required to be in service to allow the release to continue. The 1R18 does not isolate or close the 1WL51 on loss of flow.

B. Incorrect. Plausible because the operator may believe that you need both circulators in service to continue the discharge.

C. Incorrect. Plausible because the operator may believe that the 1R18 radiation monitor will trip the 1WL51 closed when loss of flow is detected or may believe the operator may believe that you need both circulators in service to continue the discharge.

D. Incorrect. Plausible because the operator may believe that the 1R18 radiation monitor will trip the 1WL51 closed when loss of flow is detected.

Technical References:	S1.OP-SO.WL-0001 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05WASLIQ-11, Objective 12. Discuss the procedural requirements associated with the Radioactive Liquid Waste System, including an explanation of major precaution and limitations in the Radioactive Liquid Waste System procedures.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.3
Level:	RO
Category/System:	08-Liquid Radwaste System
K/A #:	068 K5.07 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Liquid Radwaste System: Loss of secondary mixing water system
Importance:	2.6
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the applicant is being asked how the loss of one circulator will affect the liquid release in progress.

62 **07-028** **Points: 1.00**

Given:

- Salem Units 1 and 2 are at 100% power

Then:

- 1R1A, Control Room Area, Radiation Monitor is in Alarm

Which of the following actions is required?

Note: CAV = Control Room Area Ventilation
AP = Accident Pressurized

A. Verify automatic actuation of ONLY Unit 1 CAV in AP Mode.

B. Verify automatic actuation of ONLY Unit 2 CAV in AP Mode.

C. Manually press AP pushbutton at 1RP2 panel to actuate BOTH units CAV in AP Mode.

D. Manually press AP pushbutton at 2RP2 panel to actuate BOTH units CAV in AP Mode.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may confuse the 1R1A with the 1R1B radiation monitors and believe that it will actuate CAV in AP Mode only on the affected unit. Incorrect in that 1R1A area radiation monitors will not actuate CAV in AP Mode.

B. Incorrect. Plausible because the operator may confuse the 1R1A with the 1R1B radiation monitors and believe that it will actuate CAV in AP Mode only on the affected unit. Incorrect in that 1R1A area radiation monitors will not actuate CAV in AP Mode.

C. Correct. Per the ARP for 1R1A radiation monitor in alarm, the control room crew will be required to place the Control Room Ventilation (CAV) System into Accident Pressurized (AP) Mode. Since the area radiation monitor is alarm sensing radiation on Salem Unit 1 side, the crew will actuate CAV in AP Mode from Unit 1 which will actuate both units CAV into AP Mode and open the emergency intake dampers on the unaffected unit, in this case unit 2.

D. Incorrect. Plausible because the operator may confuse which unit to actuate CAV in AP Mode to align the emergency intake dampers on the unaffected unit, in this case unit 2.

Technical References:	S2.OP-AB.RAD-0001 (R30), S2.OP-SO.CAV-0001 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05REFUEL-13,
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.7 / 45.5 to 45.8
Level:	RO
Category/System:	07-Area Radiation Monitoring System
K/A #:	072 A4.02 Ability to manually operate and/or monitor the Area Radiation Monitoring System in the control room: Radiation monitor function
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking what action is required when the control room area radiation monitor goes into alarm.

Additional Reference:

S1.OP-AB.RAD-0001(Q)

ATTACHMENT 3
(Page 1 of 1)

AREA RADIATION MONITORS

___ 1.0 IF 1R1A, Control Room Area, is the affected monitor, THEN:

___ 1.1 Coordinate with Unit 2 NCO to **ALIGN** Control Room Ventilation for ACCIDENT PRESSURIZED mode IAW the following procedures:

- S1.OP-SO.CAV-0001(Q), Control Room Ventilation Operation
- S2.OP-SO.CAV-0001(Q), Control Room Ventilation Operation

___ 1.2 DIRECT Radiation Protection to survey area to determine source of activity rise.

Additional Reference:

5.7 Manual Initiation of ACCIDENT PRESSURIZED Mode Operation

NOTE

___ This Section provides direction for placing both Units in the ACCIDENT PRESSURIZED mode manually initiated by Unit 1. If ACCIDENT PRESSURIZED mode was initiated from Unit 2, Step 5.7.1 should NOT to be performed.

___ 5.7.1 **PRESS** ACCIDENT PRESSURIZED pushbutton at 1RP2.

NOTE

___ IF ACCIDENT PRESSURIZED mode was initiated from Unit 2,
THEN NORMAL lamp will remain illuminated.

___ 5.7.2 **ENSURE** red ACCIDENT PRESSURIZED lamp illuminates.

___ 5.7.3 **ENSURE** CONTROL AREA ISOLATION Train A ISOLATION PUSH TO RESET red lamp is illuminated AND Train B ISOLATION PUSH TO RESET red lamp is illuminated.

___ 5.7.4 IF an accident has occurred on Unit 1,
THEN **NOTIFY** Unit 2 NCO to PERFORM S2.OP-SO.CAV-0001(Q), Control Area Ventilation Operation, section on Automatic Initiation of ACCIDENT PRESSURIZED Mode Operation.

___ 5.7.5 IF manual initiation of ACCIDENT PRESSURIZED mode operation is NOT initiated due to an accident,
THEN **NOTIFY** Unit 2 NCO to PERFORM the following:

___ A. **ENSURE** Unit 2 CAACS is aligned to ACCIDENT PRESSURIZED mode of operation IAW S2.OP-SO.CAV-0001(Q), Control Area Ventilation Operation, Exhibit 1 for Component Positions.

63 **08-024** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- OHA A-7, FIRE PROT FIRE, is alarming
- Fire alarm zones on 2RP5 Panel indicates a fire detected inside Panel 335 in containment

Which of the following completes the following statement concerning the operation of the 2FP147, containment fire header isolation valve?

1. Based on a valid fire detected inside containment, the 2FP147 will __(1)__.
2. On a Phase A signal, the 2FP147 __(2)__ automatically close.

A. (1) automatically open
(2) will NOT

B. (1) automatically open
(2) will

C. (1) need to be manually opened from the 2RP5 panel
(2) will

D. (1) need to be manually opened from the 2RP5 panel
(2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because fire protection systems will actuate if the fire detector sensor is triggered. Incorrect in that even though the fire zone fire water header is triggered, the 2FP147 still needs to be manually opened from the control room.

B. Incorrect. Plausible because fire protection systems will actuate if the fire detector sensor is triggered. Incorrect in that even though the fire zone fire water header is triggered, the 2FP147 still needs to be manually opened from the control room.

C. Correct. The 2FP147 requires to be manually opened at the 2RP5 panel when a valid fire is detected inside containment. The 2FP147 will close on a Phase A signal.

D. Incorrect. Plausible because the operator may believe that fire water inside containment is the priority and will not auto isolate.

Technical References:	NOS05FIRPRO-12
Proposed References to be provided:	None
Learning Objective:	NOS05FIRPRO-12, Objective 13. Given plant conditions, relate the Fire Protection System with the following: Containment and Containment Press Relief
Question Source:	Modified – Salem 2019 NRC RO75
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 45.5
Level:	RO
Category/System:	08-Fire Protection System
K/A #:	086 A1.06 Ability to predict and/or monitor changes in parameters associated with operating the Fire Protection System, including: Lights and alarms
Importance:	3.0
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the question is asking the applicant how the fire water is supplied into containment when a valid fire is detected. The operator requires knowledge of hoe the 2FP147 fire header isolation valve functions during a valid fire detected and whether the 2FP147 will isolate on a Phase A containment isolation signal..

Additional Reference:

ALARM **A-7**	7 FIRE PROT FIRE

DEVICES:	N/A
SETPOINT:	N/A
REFERENCES:	Dwg. 203753, 205447, 601270, 203750, 203751

A-7 FIRE PROT FIRE

1.0 CAUSE(S):

1.1 Any:

A. Coded fire alarm

B. Fire alarm on 2RP5

2.0 AUTOMATIC ACTION:

NONE

3.0 OPERATOR ACTIONS:

3.1 **SCAN** 2RP5 to determine Fire Zone, Area, and Elevation
OR **DETERMINE** location IAW Alarm Code List.

3.2 IF AT ANY TIME, fire indication for both Zones 59 and 74 on 2RP5 are received,
THEN OPEN 2FP147, Fire Protection Containment Isolation.

Parent Question: Salem 2019 NRC RO75

Modified question based on the stem asking about isolation on a Phase A.

Question: #75

Given:

- Unit 2 is operating at 100 % Power.
- OHA A-7, FIRE PROT FIRE, annunciates.
- Panel 2RP5 is checked and indicates a Fire in Containment.
  - Zone 59 – Air and Water Deluge, Containment El. 100 Panel 335 is lit.
  - Zone 74 – Smoke and Fire Detector, Containment El. 100 Panel 335 is lit.
- The crew enters S2.OP-AB.FIRE-0001, Control Room Fire Response.
- The crew Trips the Reactor, Turbine, and all RCPs.
- 2-EOP-TRIP-1 is entered while continuing with S2.OP-AB.FIRE-0001, Control Room Fire Response.

Which ONE of the following describes the **NEXT** required action for the above conditions in accordance with S2.OP-AB.FIRE-0001?

A. OPEN the 2FP147 from the control room

B. Dispatch an NEO to OPEN the associated deluge valves.

C. Dispatch an NEO to place both PORV BLOCK Valve breaker key switches in EMER CLOSE.

D. Verify OHA A-15, FIRE PUMP 1/2 RUN, is in alarm indicating that a Diesel Fire Pump has started and is supplying fire protection water to the associated deluge valves in containment.

Answer: A

64 **14-087** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- All systems are in normal alignment and in Auto

Then:

- 21 SGFP trips

Which of the following indications is the expected response?

1. The DEHC will automatically runback the turbine at __(1)__ per minute.
2. 2CN47 __(2)__ immediately open when the feedpump trips.

A. (1) 15%
(2) will NOT

B. (1) 15%
(2) will

C. (1) 200%
(2) will

D. (1) 200%
(2) will NOT

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because this is the ramp rate when manually initiating a turbine runback at the DEHC panel. Plausible because on a condensate pump trip, the 2CN47 will only open if SGFP suction pressure lowers to less than 320 psig. Incorrect in that the 2CN47 will open following a SGFP trip regardless of suction pressure.

B. Incorrect. Plausible because this is the ramp rate when manually initiating a turbine runback at the DEHC panel.

C. Correct. A SGFP will automatically initiate a turbine runback at 200% per minute to 66% power. When a SGFP trips the 2CN47 will open and the polisher will be bypassed.

D. Incorrect. Plausible because on a condensate pump trip, the 2CN47 will only open if SGFP suction pressure lowers to less than 320 psig. Incorrect in that the 2CN47 will open following a SGFP trip regardless of suction pressure.

Technical References:	S2.OP-AB.CN-0001 (R32)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCN01-07, Objective 1.d. Describe the operation of the following system as applied to AB.CN-0001: SGFP Trip
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.19 Ability to use available indications to evaluate system or component status.
Importance:	3.9
Tier/Group	General Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked the expected indications following a SGFP trip.

65 **14-088** **Points: 1.00**

Given:

- Salem Unit 1 experienced a valid demand for a reactor trip and Safety Injection (SI)
- The RO suspects that only one Train of SI actuated

Which of the following indications would corroborate that one train of SI failed to actuate?

The "SI & FW ISOL" Red lamp on 2RP4 panel will be __(1)__ and the "SI RESET" Green bezel light on the failed Train of Safeguards on 2CC1 console will be __(2)__.

A. (1) Flashing
(2) LIT

B. (1) Flashing
(2) Extinguished

C. (1) LIT
(2) Extinguished

D. (1) LIT
(2) LIT

Answer: A

Explanation / Justification:

A. Correct. The "SI & FW ISOL" Red lamp on 2RP4 would be flashing indicating SSPS train disagreement and that one Train of SI has failed to actuate. Also, the "SI REST" Green light on the train that failed to actuate will be LIT

B. Incorrect. Plausible because the "SI & FW ISOL" Red lamp on 2RP4 would be flashing indicating SSPS train disagreement and that one Train of SI has failed to actuate. The SI RESET light would be extinguished but for the Train that did actuate

C. Incorrect. Plausible because "SI & FW ISOL" Red lamp on 2RP4 would be LIT solid if both trains of SI did actuate and the SI RESET light would also be extinguished for the train that SI actuated on.

D. Incorrect. Plausible because "SI & FW ISOL" Red lamp on 2RP4 would be LIT solid if both trains of SI did actuate and the SI RESET light would be LIT for the train that SI did not actuated on.

Technical References:	NOS05RXPROT-15, 218489
Proposed References to be provided:	None
Learning Objective:	NOS05RXPROT-15, Objectives 19a & b. Identify and describe the Control Room controls, indications, and alarms associated with the Reactor Protection System
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 45.12
Level:	RO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.31 Ability to locate control room switches, controls, and indications and to determine whether they correctly reflect the desired plant lineup.
Importance:	4.6
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked to determine how to validate if one train of SI has actuated and the location of the indication. In this case, the applicant must know the indication in the stem is located on 2RP4 and should be flashing to indicate train disagreement,

Additional Reference: Excerpt from NOS05RXPROT

C. RP4
   1. Indication only
   2. Used to verify a reactor trip demand
   3. Used to verify safeguards equipment status
   4. If light flashes, Train A & B do not agree. One multiplexer switch must be in "A + B" for this to occur.
      a. This Train Disagreement feature of RP4 applies to all the bistable indicating lights, plus the blue AUTO SI BLOCK light.
      b. It does NOT apply to the blue block lights associated with the blockable reactor trips (for example, SR Hi Flux) and the blockable SI's (for example, Low PZR Pressure). For these functions, RP4 shows "Train B" as an input to the 2/2 logic function – in other words – it takes both Train A and Train B to illuminate the light, and there is no flashing feature.

D. General Warning Alarm System
   1. Warns operator if SSPS is not entirely operational
   2. Illuminates a light on logic bay panel

66 **08-025** **Points: 1.00**

Which of the following completes the statement concerning Salem Unit 1 vs. Unit 2 Stator Water Cooling System design?

Salem Unit __(1)__ is the ONLY unit that has an automatic Turbine Runback feature when the following stator water system condition(s) exist; __(2)__.

A. (1) 1
(2) low flow, low pressure, or high temperature

B. (1) 2
(2) low flow, low pressure, or high temperature

C. (1) 2
(2) low flow Only

D. (1) 1
(2) low flow Only

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse which unit has this design feature. Part 2 is correct.

B. Correct. Salem Unit 2 is the only unit that has an automatic turbine runback when any of the following conditions exist; low flow, low pressure, and high temperature.

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because this condition is used on both units, but unit 2 uses it for the turbine runback whereas unit 1 uses it for a turbine trip feature only.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse which unit has this design feature. Part 2 is incorrect. Plausible because this condition is used on both units, but unit 2 uses it for the turbine runback whereas unit 1 uses it for a turbine trip feature only.

Technical References:	NOS05SWCU02-09
Proposed References to be provided:	None
Learning Objective:	NOS05SWCU02-09, Objective 11. Identify the differences between Unit 1 and Unit 2 Stator Cooling Water System components, parameters, and operation
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.5 / 41.6 / 41.7 / 41.10 / 45.12
Level:	RO
Category/System:	14-Equipment Control
K/A #:	G2.2.3 Knowledge of the design, procedure, and/or operational differences between units.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked the design difference between the units stator water cooling system. In this case, the applicant needs to know that only unit 2 has a automatic turbine runback feature.

Additional References: NOS05SWCU02

V. UNITS 1 & 2 DIFFERENCE

A. Unit 2 is a General Electric SCW Unit. Loss of flow, low pressure and hi temperature are inputs to the turbine runback circuit. If runback fails to lower amps to the specified value within the specified time delay then a turbine (generator) trip will occur.

B. Unit 1:

1. Is a Westinghouse SCW Unit (covered in a separate lesson plan).
2. Low flow inputs into a 45 second time delayed turbine trip circuit.
3. Does not have a SLMS System.

67 **14-089** **Points: 1.00**

Given:

- A valve lineup is being performed on a system that requires Independent Verification
- The valve lineup contains a valve that is required to be throttled open two turns
- Permission has been granted to move the valve as required to verify its position

What are the actions necessary to verify the position of this throttled open valve?

A. One operator will go out ALONE and fully close the valve, then reopen it two full turns. Then, the second operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

B. One operator will go out ALONE and fully open the valve, then close the valve the same number turns. Then, the second operator will go out ALONE and verify the valve is properly positioned by visual observation of the valve stem.

C. One operator will go out ACCOMPANIED by a second operator. One of the operators will fully close the valve, then reopen it two full turns while the second operator observe the actions.

D. One operator will go out ACCOMPANIED by a second operator. One of the operators will fully open the valve, then close the valve the same number of turns while the second operator observe the actions

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because typical independent verification of valve positions is performed after the first operator performs the valve manipulation and the second operator goes out alone to independently verify its position.

B. Incorrect. Plausible because typical independent verification of valve positions is performed after the first operator performs the valve manipulation and the second operator goes out alone to independently verify its position.

C. Correct. Per OP-AA-108-101-1001 and OP-AA-1008-101-1002, when a throttled valve position requires verification, then the valve is fully closed, then reopened to the required number of turns. The independent verification is performed by a second operator by either; independently counting the number of turns, verification of process variable, or verification of valve position for valves with incremental indicators against the tagout footnote or procedure.

D. Incorrect. Plausible because the operator may confuse the direction of how the throttled valve must be positioned prior to returning to its throttled position. Fully closed vs. fully open.

Technical References:	OP-AA-108-101-1001 (R2), OP-AA-101-1002 (R11)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures: Shift Relief and Turnover
Question Source:	Bank – Millstone U2 NRC RO69
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.3 / 45.13
Level:	RO
Category/System:	14-Equipment Control
K/A #:	G2.2.14 Knowledge of the process for controlling equipment configuration or status.
Importance:	3.9
Tier/Group	General Knowledge and Abilities-Tier 4

K/A Match: The K/A is matched because the candidate is being asked how to verify a throttled open valve position and independent verification.

Additional References: OP-AA-108-101-1001

4.4 Checking Manual Throttle Valve Position

4.4.1. Valves that are required to be placed in a throttled position will normally be checked in the proper position by using the appropriate process parameter (flow, pressure, etc.).

4.4.2. When operation of a throttled valve is necessary to determine its position, **CONSULT** with the appropriate Operations supervisor prior to manipulating the throttled valve for guidance, including discussion with cognizant Engineering personnel as necessary.

NOTE Personnel must be aware of the effect that mechanical valve operator slack will have on the number of hand-wheel turns required to position a valve. Compensate as necessary to allow for the slack.

1. Provided system configuration allows valve movement, check throttled valve position by counting the number of 360-degree turns required to fully close the valve.

2. **RE-POSITION** the valve to the throttled position by opening the valve the same number of turns counted in the previous step. Restoring locked throttled valves requires concurrent verification.

Additional Reference: OP-AA-108-101-1002

ATTACHMENT 5
INDEPENDENT/CONCURRENT VERIFICATION
Page 3 of 4

2.4.3. For Hope Creek valves BC-HV-F077, BC-HV-F060A, BC-HV-F060B valves should be verified "open" by observation of the valve local position indication or by use of the valve remote position indication and verified "locked" by observation that a Locking Wire is Secured Between the Valve Yoke and the Wire Loop Installed on Valve Actuator Stem. **[DCP 80072763]**

2.5 When a manual valve is to be verified closed, move the handwheel in the close direction only. Do not exceed normal closing torque. **[CD-095B, CD-088B]**

NOTE

Valves that are binding on the backseat will give a false indication of being closed. **[CD-769F]**

Valves with Safety Tags are controlled by the tagging process and do not require verification.

2.6 When a manual valve is to be verified open move the handwheel in the closed direction enough to verify valve movement. Then return the valve to the open position. **[CD-095B, CD-088B]**

2.7 Throttle valves:

2.7.1. Initial alignment or verification:

1. Fully close the valve.
2. Opened the valve the required number of turns, or if applicable, match the alignment marks.

2.7.2. Independent verification, the second operator may observe the first operator positioning the valve, then perform either an independent: **[CD-286B]**

- Count of the turns open.
- Verification of process variable.
- Verification of valve position for valves with incremental indicators against the WCM footnote or procedure.

68 **14-090** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- Release of 21 CVCS Monitor Tank is in progress per S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste from 21 CVCS Monitor Tank

Then:

- 2FR1064, Radwaste Overboard Discharge Flow Recorder, fails low

Per S2.OP-SO.WL-0001, which of the following completes both statements?

1. Based on the above condition, the 2WL51, Overboard Valve, __(1)__ automatically close..
2. With 2FR1064 Flow recorder inoperable, the release __(2)__.

A. (1) will
(2) may continue if effluent flows are estimated once per 4 hours

B. (1) will NOT
(2) may NOT recommence unless 2FR1064 is OPERABLE

C. (1) will
(2) may NOT recommence unless 2FR1064 is OPERABLE

D. (1) will NOT
(2) may continue if effluent flows are estimated once per 4 hours

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that a low flow condition will automatically close the 2WL51. Incorrect in that loss of flow is not an enabled design feature with the radiation monitor.

B. Incorrect. Plausible because the operator may believe that the procedure requires both 2R18 and 2FR1064 operable in order to proceed with the release and satisfy the requirements of the ODCM.

C. Incorrect. Plausible because the operator may believe that a low flow condition will automatically close the 2WL51. Plausible because the operator may believe that the procedure requires both 2R18 and 2FR1064 operable in order to proceed with the release and satisfy the requirements of the ODCM.

D. Correct. Loss of discharge flow is not a design feature associated with the 2R18 process radiation monitor. Some process radiation monitors do have a feature that when programmed will isolate the process on low of flow. Per ODCM bases for liquid effluent (3.3.8), the controls and monitoring are determined to ensure that the 10 CFR 20 limits are not exceeded. Per S2.OP-WL-0001 and Action 29 of the ODCM for an inoperable liquid radwaste effluent line monitor, the release may continue provided effluent flow is estimated once per 4 hours.

Technical References:	ODCM (R29), S2.OP-SO.WL-0001 (R32)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures
Question Source:	Bank – Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.11 / 43.4 / 45.10
Level:	RO
Category/System:	14-Radiation Control
K/A #:	G2.3.11 Ability to control radiation releases.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate is being asked the bases of the ODCM administrative limits for the release of liquid radwaste and whether the release may proceed if actions are taken per the procedure/ODCM.

Additional References: Salem ODCM 3.3.8

PART I RADIOLOGICAL EFFLUENT CONTROLS — Salem REC ODCM Rev. 29 03/21

3/4.3 INSTRUMENTATION

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE 3.3-12: Radioactive Liquid Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Liquid Radwaste Effluent Line	1 (Unit 1) - **1R18** 1 (Unit 2) - **2R18**	26
b. Steam Generator Blowdown Line	4 (Unit 1) - **1R19A-D** 4 (Unit 2) - **2R19A-D**	27
2. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Containment Fan Coolers – Service Water Line Discharge	2 (Unit 1) - **1R13A, B** 2 (Unit 2) - **2R13A, B**	28
b. Chemical Waste Basin	1 **(R37)**	31
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1 (Unit 1) - **1FR1064** 1 (Unit 2) - **2FR1064**	29
b. Steam Generator Blowdown Line	4 (Unit 1) - **1FA-3178, -3180, -3182, -3184** 4 (Unit 2) - **2FA-3178, -3180, -3182, -3184**	29

Additional References: Salem ODCM 3.3.8 Bases

BASES

3/4.3 INSTRUMENTATION

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of the "old" 10 CFR Part 20 (ODCM Appendix F). The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

CROSS REFERENCE - TABLES 3.3-12 and 4.3-12

Unit 1 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Liquid Radwaste Effluent Line Gross Activity	1R18
1b	Steam Generator Blowdown Line Gross Activity	1R19A, B, C, and D(1)
2a	Containment Fan Coolers Service Water Line Discharge Gross Activity	1R13A and B(1)

Unit 2 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Liquid Radwaste Effluent Line Gross Activity	2R18
1b	Steam Generator Blowdown Line Gross Activity	2R19A,B,C, and D(1)
2a	Containment Fan Coolers - Service Water Line Discharge Gross Activity	2R13A and B(1)
2b	Chemical Waste Basin Line Gross Activity	R37

(1) The channels listed are required to be operable to meet a single operable channel for the ODCM's "Minimum Channels Operable" requirement.

69 **14-091** **Points: 1.00**

Given:

- You are the Primary Communicator

Then:

- A station event has just been classified as a Site Area Emergency

Which of the following describes the time requirement for State and Local notifications?

A. 30 minutes from the initiation of the event.

B. 15 minutes from the completion of the initial contact message form.

C. 15 minutes from the time the communicator acknowledges the briefing from the Shift Manager.

D. 15 minutes from the classification of the event.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the notification time includes the 15 minutes to classify and 15 minutes for the notification to the States. Incorrect in that the classification time can be less than 15 minutes, therefore 30 minutes for notifications is not always correct.

B. Incorrect. Plausible because the operator as the primary communicator usually waits until the ICMF is completed by the Shift Manager and is read to the communicator before notifying the States. Incorrect in that the notification time of 15 minutes starts when the Shift Manager declares the emergency classification.

C. Incorrect. Plausible because the operator as the primary communicator usually waits until the ICMF is completed by the Shift Manager and is read to the communicator before notifying the States. Incorrect in that the notification time of 15 minutes starts when the Shift Manager declares the emergency classification.

D. Correct. The 15 minute notification time to the States and Local agencies starts when the Shift Manager classifies or declares the emergency classification for the event.

Technical References:	EP-SA-325-F6 (R16)
Proposed References to be provided:	None
Learning Objective:	None
Question Source:	Bank – Beaver Valley 2017 RO75
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.3 / 45.11
Level:	RO
Category/System:	14-Emergency Procedures/Plan
K/A #:	G2.4.29 Knowledge of the emergency plan implementing procedures.
Importance:	3.1
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the applicant is being asked about Primary Communicator duties for time requirements to notify the States and Local agencies of the emergency classification.

70 **16-018** **Points: 1.00**

Given:

- Salem Unit 1 has been operating at 100% power for the past six months

Then:

- The reactor trips
- Shortly afterwards, a reactor startup is performed
- Twelve (12) hours after the trip, the startup has been paused with reactor power at 5%

To maintain reactor power and Tavg stable over the next hour, the operator must add __(1)__ reactivity because the xenon-135 concentration will be __(2)__?

A. (1) negative (2) rising

B. (1) negative (2) lowering

C. (1) positive (2) lowering

D. (1) positive (2) rising

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a higher xenon concentration will require negative reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity..

B. Correct. Directly following a reactor trip, the neutron flux is reduced essentially to zero. Therefore, Xe-135 is no longer produced directly from fission, but is no longer removed by burnup. The only remaining production mechanism is β-decay of the iodine-135; the only removal mechanism for xenon-135 is β-decay. Because the decay rate of iodine-135 is faster than the decay rate of xenon-135, the concentration builds to a peak (about -4700 pcm if tripped at 100% power). The time to peak is slightly less than the square root of the power from the trip (8-9 hours from 100%). Following the peak, the Xe-135 concentration will decrease at a rate controlled by the decay of Xe-135. Therefore, an additional twelve (12) hours after the power change, Xe-135 concentration will be decreasing due to the decay of I-135 and Xe-135. Additionally, returning to 5% power will result in a small amount of the burnout term removing Xe-135. A reduction in Xe-135 inserts positive reactivity (Xe-135 is a fission product poison). Therefore, to maintain s table reactor coolant temperature, the operator must insert negative reactivity because xenon-135 concentration is decreasing.

C. Incorrect. Part 1 is incorrect. Part 2 is correct. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a lower xenon concentration will require adding positive reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity.

D. Incorrect. Part 1 is incorrect. Part 2 is incorrect. Plausible if the operator confuses the concept on how the fission product poison (xenon-135) effects reactivity and believes that a higher xenon concentration will require positive reactivity. Incorrect in that after 12 hours xenon has already peaked and xenon concentration will be decreasing adding positive reactivity, and therefore, require adding negative reactivity.

Technical References:	INPO GFES Fission Product Poisons (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05POISON-02. INPO GFES Fission Product Poisons Rev. 3, ELO 1.4 Xenon Terms
Question Source:	Bank – NRC GFES Bank P3563
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactor Theory / Fission Product Poison
K/A #:	192006 K1.07 Describe the following processes and state their effect on reactor operations – xenon following a trip.
Importance:	3.4
Tier/Group	Theory-Tier 4

K/A Match: The K/A is a match because the candidate is being asked knowledge of how xenon-135 concentration is effecting reactivity following a reactor trip and what action the operator will need to take to compensate for the xenon.

Additional Reference: INPO GFES Fission Product Poison – Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P3563**
	192006 – FISSION PRODUCT POISONS	**K/A: K1.14 [3.2/3.3]**

A nuclear power plant had been operating at 100 percent power for two months when a reactor trip occurred. Soon afterward, a reactor startup was performed. Twelve hours after the trip, the startup has been paused with reactor power at 5 percent.

To maintain reactor power and reactor coolant temperature stable over the next hour, the operator must add _____ reactivity because the xenon-135 concentration will be _____.

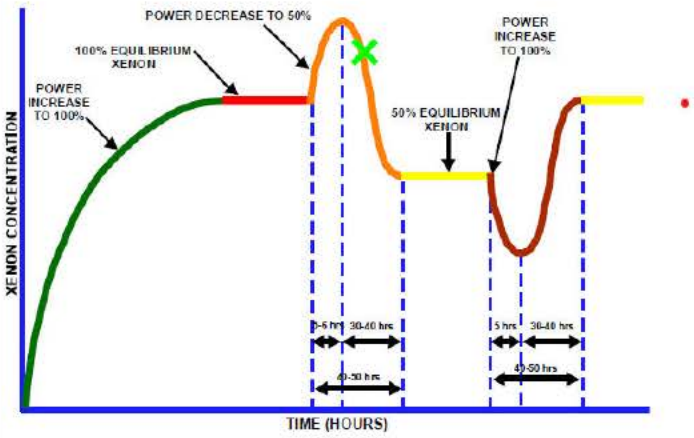
A. X positive; X increasing.

B. √ negative; X increasing.

C. X positive; √ decreasing.

D. √ negative; √ decreasing.

Discussion/Calculation:



Directly following a reactor trip, the neutron flux is reduced essentially to zero. Therefore, Xe-135 is no longer produced directly from fission, but is no longer removed by burnup. The only remaining production mechanism is the β- decay of the iodine-135; the only removal mechanism for xenon-135 is β- decay. Because the decay rate of iodine-135 is faster than the decay rate of xenon-135, the xenon concentration builds to a peak (about -4700 pcm if tripped from 100% power).

The time to "peak" is slightly less than the square root of the power from the trip (8-9 hours from 100% power). Following the peak, the Xe-135 concentration will decrease at a rate controlled by the decay of Xe-135.

Therefore, an additional twelve (12) hours after the power change, Xe-135 concentration will be *decreasing* due to the decay of I-135 and Xe-135. Additionally, returning to 5% power will result in a small amount of the burnout term removing Xe-135. A reduction in Xe-135 inventory inserts positive reactivity (Xe-135 is a fission product poison). *Therefore, to maintain a stable reactor coolant temperature, the operator must insert **negative** reactivity because xenon-135 concentration is **decreasing**.*

71 **16-019** **Points: 1.00**

Given:

- Salem Unit 2 was operating at 100% power when a Loss of All Offsite Power occurs.
- Due to plant complications, the Shift Manager has directed the crew to cooldown the RCS to Cold Shutdown per 2-EOP-TRIP-4, Natural Circulation Cooldown.
- The crew is borating the RCS for the RCS cooldown to Cold Shutdown.

Which of the following completes both statements?

1. As compared to adding boron to the RCS during forced circulation, adding boron during natural circulation requires __(1)__ time to achieve complete mixing in the RCS.
2. Once the RCS boron mixing is complete, a 1 ppm increase in RCS boron concentration during natural circulation will cause a/an __(2)__ change in reactivity that it would during forced circulation operation.

. A. (1) less
(2) smaller

B. (1) less
(2) equal

C. (1) more
(2) smaller

D. (1) more
(2) equal

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible if the operator misreads the question and therefore believes the question is asking about time to mix when forced circulation. Part 2 is incorrect. Plausible if the operator believes that reactivity effects will be smaller due to the lower RCS flowrates. Incorrect in that the change in reactivity will be the same when the boron is thoroughly mixed.

B. Incorrect. Part 1 is incorrect. Part 2 is correct. Plausible if the operator misreads the question and therefore believes the question is asking about time to mix when forced circulation

C. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible if the operator believes that reactivity effects will be smaller due to the lower RCS flowrates. Incorrect in that the change in reactivity will be the same when the boron is thoroughly mixed.

D. Correct. Due to loss of offsite power, no RCPs will be running resulting in a significantly lower RCS flowrate (NC conditions). Therefore, more time will be required to achieve complete mixing in the RCS for a given boron concentration change. Once the RCS mixing is complete, a 1 ppm increase (or decrease) in RCS boron concentration during NC operation will cause the same change in core reactivity that it would during forced circulation operation.

Technical References:	INPO GFES Fuel Depletion and Burnable Poisons (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05POISON-02. Refer to INPO GFES Fuel Depletion and Burnable Poisons Rev. 3, ELO 1.6 Boron and Natural Circulation
Question Source:	Bank – NRC GFES Reactor Theory P3364
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactory Theory / Fuel Depletion and Burnable Poisons
K/A #:	192007 K1.05 Describe the effects of boration/dilution on reactivity during forced-flow and natural circulation conditions.
Importance:	3.2
Tier/Group	Theory-Tier 4

K/A Match: The K/A is a match because the candidate is being asked how boration is effected during natural circulation.

Additional Reference: INPO GFES Fuel Depletion and Burnable Poisons

ELO 1.6 Boron and Natural Circulation

Introduction

This section contains a short discussion of boron concentration changes during natural circulation conditions.

Boron Concentration Changes During Natural Circulation

Normally, boration and dilution are not performed during natural circulation conditions. Without the Reactor Coolant Pumps (RCPs) running the RCS flow rate will be much lower and the boron mixture distribution will be uneven throughout the core. This could lead to a loss of shutdown margin. To prevent a dilution accident, reducing boron concentration during natural circulation conditions is not normally allowed. Any boron changes in natural circulation will have the same reactivity effect as with forced circulation, once completely mixed.

The concentration of boron used for RCS boration remains the same throughout core life (from BOL to EOL) so the rate at which negative reactivity is added during boration does not change over core life. However, because the concentration of boron in the RCS decreases over core life by approximately a factor of 10, the rate at which dilution occurs (and thus, reactivity is added) is much slower (by approximately a factor of 10 as well).

So even though the reactor will respond quicker at the end of cycle (EOC) for a given reactivity positive addition from a boron dilution, the reactivity can be added quicker at the beginning of cycle (BOC) due to quicker changes in boron concentration.

72 **16-020** **Points: 1.00**

Given:

- Salem Unit 1 is performing a reactor startup with the reactor slightly subcritical.
- 2N31 and 2N32 Source Range count rates are stable.

Then:

- The RO performs a short control rod withdrawal.

Which of the following describes the response on the Source Range (SR) counts and startup rate after the rod withdrawal has stopped (assume the reactor remains subcritical)?

A. SR counts will slowly increase then gradually increase and stabilize at a higher value; startup rate will rapidly increase then gradually decrease to a slightly positive value.

B. SR counts will rapidly increase then stabilize at a higher value; startup rate will rapidly increase then gradually decrease and stabilize at zero.

C. SR counts will rapidly increase then gradually decrease and stabilize at the previous value; startup rate will rapidly increase then gradually decrease and stabilize at zero.

D. SR counts will rapidly increase then stabilize at a higher value; startup rate will not change until criticality is achieved.

Answer: B

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may believe that the SR counts will continue to increase sometime after the rod pull has stopped. Incorrect in that this describes the SR count rate when the reactor is near critical conditions. Second part is incorrect. Plausible because the operator may believe that a stable positive SUR is necessary to achieve a higher equilibrium SR count rate. Incorrect in that in subcritical multiplication, SUR will initially rise then decay to zero with a new stable SR count rate.

B. Correct. The reactor will have a stable count rate in the SR while subcritical (keff < 1.0). When the operator stops rod motion in a subcritical reactor, the source range count rate will achieve a new higher equilibrium level due to the insertion of positive reactivity (keff increases, but still not critical). With each rod pull positive reactivity is inserted and the startup rate will increase then decay to zero indicating subcritical multiplication has reached equilibrium.

C. Incorrect. Plausible because the operator may believe that if the SUR is not constant, then the SR count rate will not achieve a higher value.

D. Incorrect. Plausible if the operator believes that the SR startup rate will not change until the reactor is critical.

Technical References:	INPO Reactor Operational Physics (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05RXOPER-02. Refer to INPO Reactor Operational Physics Rev. 3, ELO 1.2 Reactor Startup Nuclear Instrumentation Response.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.1
Level:	RO
Category/System:	16-Reactory Theory / Reactor Operational Physics (Startup and Approach to Criticality)
K/A #:	192008 K1.03 Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality.
Importance:	4.0
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked how the SR count rate and startup rate behave during a subcritical reactor during a reactor startup.

Additional Reference: INPO GFES Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P265**
	192008 – REACTOR OPERATIONAL PHYSICS	**K/A: K1.03 [3.9/4.0]**

A reactor startup is in progress and the reactor is slightly subcritical in the source range. Assuming the reactor remains subcritical, a short control rod withdrawal will cause the reactor startup rate indication to increase sharply in the positive direction, and then...

A. X rapidly decrease and stabilize at a negative 1/3 dpm.

B. √ gradually decrease and stabilize at zero.

C. X stabilize until the point of adding heat (POAH) is reached; then decrease to zero.

D. X continue increasing until the POAH is reached; then decrease to zero.

Discussion/Calculation:

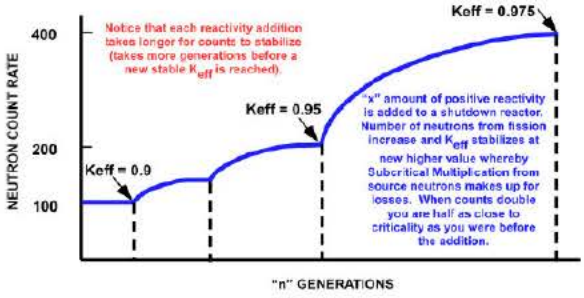
A reactor will have a stable count rate in the source range while subcritical (k_{eff} < 1.0). Because k_{eff} < 1.0, each neutron generation will produce less fission neutrons than the previous generation; source neutrons make up for the losses each generation in fission neutrons to stabilize the count rate. *Subcritical multiplication* is the process of utilizing source neutrons and fuel (fission neutrons) to maintain constant neutron population with k_{eff} less than 1.0. If positive reactivity is inserted in the core (k_{eff} increases, but still not critical) the source range count rate will increase:

$$N_t = S_o\left(\frac{1}{1-K_{eff}}\right) \text{ or } N_t = S_o M$$

SUR indication during a rod withdrawal in a subcritical reactor looks like this:



1 – Jump Up (when rods movement started)
2 – Smile (during continuous rod withdrawal)
3 – Jump Down (when rod movement stopped)
4 – Decay to 0 as count rate stabilizes in the subcritical reactor.



Where, N_t = total neutrons, S_o = source neutrons, $1/(1-K_{eff})$ is referred to as the subcritical multiplication factor (M)

When positive reactivity is added to a subcritical reactor, an equilibrium condition occurs when fission neutron losses equal source neutron strength, causing startup rate (SUR) to initially increase then stabilize once the new steady state neutron count rate is reached (the stem indicates the reactor remains subcritical). The indication of SUR is basically the inverse of the graph above.

Additional Reference: INPO GFES Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**REACTOR THEORY**	**P1065**
	192008 – REACTOR OPERATIONAL PHYSICS	**K/A: K1.03 [3.9/4.0]**

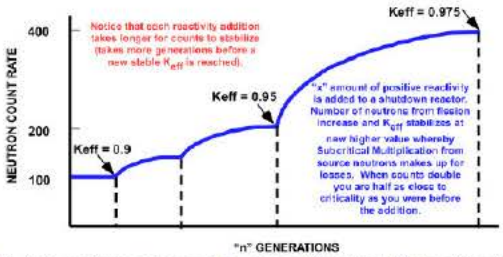
During a reactor startup, equal amounts of positive reactivity are being sequentially added, and the source range count rate is allowed to reach equilibrium after each addition. Which one of the following statements applies for each successive reactivity addition?

A. X The time required to reach equilibrium count rate is the same.
B. X The time required to reach equilibrium count rate is shorter.
C. √ The numerical change in equilibrium count rate is greater.
D. X The time required to reach equilibrium count rate is the same.

Discussion/Calculation:

A reactor will have a stable count rate in the source range while subcritical ($k_{eff} < 1.0$). Because $k_{eff} < 1.0$, each neutron generation will produce less fission neutrons than the previous generation; source neutrons make up for the losses each generation in fission neutrons to stabilize the count rate. *Subcritical multiplication* is the process of utilizing source neutrons and fuel (fission neutrons) to maintain constant neutron population with k_{eff} less than 1.0. If positive reactivity is inserted in the core (k_{eff} increases, but still not critical) the source range count rate will increase:

$$N_t = S_o\left(\frac{1}{1-K_{eff}}\right) \; or \; N_t = S_o M$$



Where, N_t = total neutrons, S_o = source neutrons, $1/(1-K_{eff})$ is referred to as the subcritical multiplication factor (M)

When positive reactivity is added to a subcritical reactor, an equilibrium condition occurs when fission neutron losses equal source neutron strength, causing startup rate (SUR) to initially increase then stabilize once the new steady state neutron count rate is reached (the stem indicates the reactor remains subcritical). Note that doubling the count rate by reactivity additions will reduce the margin to criticality by approximately one half.

It can be seen both in the equation and the graph above that as k_{eff} approaches 1.0 (criticality), the neutron population will increase at a much greater rate the closer the reactor is to criticality.

$$N_t = S_o\left(\frac{1}{1-K_{eff}\uparrow}\right) \rightarrow N_t \Uparrow$$

73 **16-021** **Points: 1.00**

Given:

- Salem Unit 2 is shutdown.
- RCS pressure is 1805 psig and stable.
- Core decay heat is being removed via the Steam Generators (SGs) using the MS10s.

What is the HIGHEST pressure in the SGs to obtain a MINIMUM 100 °F subcooling margin in the reactor coolant leaving the SGs (assume the reactor coolant leaves the SGs at the SG saturation temperature)?

A. 790 psig

B. 805 psig

C. 820 psig

D. 835 psig

Answer: B

Explanation / Justification:

A. Incorrect. Plausible if the operator incorrectly calculates the required subcooling temperature or incorrectly reads the Steam Table value.

B. Correct. At 1805 psig or 1820 psia the RCS saturation temperature is 623 °F. Therefore, to obtain a 100 °F subcooling margin in the RCS loop cold legs, temperature must be 100 °F lower than 623 °F (which is equal to 523 °F). Because a negligible temperature difference exist across the steam generator tubes, steam generator temperature must also be approximately 523 °F. Because core heat is being removed by the SGs, the SGs are a saturated system. Therefore, to determine correct SG pressure to obtain 100 °F subcooling margin, one must determine the saturation pressure at 523 °F, which is approximately 820 psia (which is equal to approximately 805 psig)..

C. Incorrect. Plausible if the operator fails to convert 820 psia to psig.

D. Incorrect. Plausible if the operator incorrectly calculates the required subcooling temperature or incorrectly reads the Steam Table value.

Technical References:	ASME Steam Tables
Proposed References to be provided:	ASME Steam Tables Rev. 2000
Learning Objective:	NOS05STEAM0-02.
Question Source:	Modified Bank – INPO GFES Bank P575
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Steam
K/A #:	193003 K1.24 Explain the usefulness of steam tables to the control room operator.
Importance:	3.1
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked to determine the saturation pressure in the steam generators that would allow for a 100 °.F subcooling in the RCS by using steam tables. This question shows the operators how useful steam tables can be when determining subcooling.

Parent Question:

NATIONAL ACADEMY FOR NUCLEAR TRAINING **PWR GFE SOLUTION**	**THERMODYNAMICS**	**P575**
	193003 – STEAM	**K/A: K1.17 [3.0/3.2]**

A reactor is shut down with reactor coolant system (RCS) pressure at 1,500 psia and core decay heat is being removed via the steam generators (SGs). What pressure must be maintained in the SGs to obtain a 110°F subcooling margin in the reactor coolant leaving the SGs? (Assume the reactor coolant leaves the SGs at the SG saturation temperature.)

A. X 580 psia

B. √ 600 psia

C. X 620 psia

D. X 640 psia

74 **16-022** **Points: 1.00**

Given:

- Salem Unit 2 is operating at 80% power.
- All Circulators are in service.

Which of the following completes the statement?

If condensate depression decreases from 5 °F to 2 °F, the steam cycle thermal efficiency will __(1)__ and the condensate pumps will operate __(2)__ cavitation.

A. (1) lower (2) closer to

B. (1) lower (2) farther from

C. (1) rise (2) closer to

D. (1) rise (2) farther from

Answer: C

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may believe with a lower condensate depression, the feedwater temperature will lower causing a rise in reactor power, and therefore, increase in plant efficiency. Second part is correct.

B. Incorrect. First part is incorrect. Plausible because the operator may believe with a lower condensate depression, the feedwater temperature will lower causing a rise in reactor power, and therefore, increase in plant efficiency. Second part is incorrect. Plausible because the operator may incorrectly recall how a lower condensate depression will affect suction pressure at the condensate pump and the likelihood of pump cavitation. Incorrect in that a lower condensate depression will increase the likelihood of pump cavitation.

C. Correct. Changing the condensate depression from 5 °F to 2 °F will decrease the condensate depression. If condensate depression decreases, then the condensate will be closer to saturation temperature. This produces less NPSH at the suction of the condensate pump, raising the probability of cavitation. However, because the condensate is now at a higher temperature, the steam generators must add less sensible heat to the feedwater to reach saturated conditions, increasing plant efficiency.

D. Incorrect. First Part is correct. Second Part is incorrect. Plausible because the operator may incorrectly recall how a lower condensate depression will affect suction pressure at the condensate pump and the likelihood of pump cavitation. Incorrect in that a lower condensate depression will increase the likelihood of pump cavitation.

Technical References:	INPO GFES Thermodynamic Process (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05THRMPR-03. Refer to INPO Thermodynamic Process Rev. 3, ELO 2.2 Condenser Design and Characteristics
Question Source:	Bank – INPO GFES P2576
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Thermodynamic Process
K/A #:	1930004 K1.11 Describe the process of condensate depression (subcooling) and its effect on plant operation.
Importance:	2.5
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked about the concept of condensate depression and how it impacts plant operation.

Additional Reference:

NATIONAL ACADEMY FOR NUCLEAR TRAINING PWR GFE SOLUTION	THERMODYNAMICS	P2576
	193004 – PROCESSES	K/A: K1.11 [2.4/2.5]

A nuclear power plant is operating at 80 percent power with 5ºF of condensate depression in the main condenser. If the condensate depression decreases to 2ºF, the steam cycle thermal efficiency will _____; and the condensate pumps will operate _____cavitation.

A. √ increase; √ closer to

B. √ increase; X farther from

C. X decrease; √ closer to

D. X decrease; X farther from

Discussion/Calculation:

Condensate depression is defined as the temperature difference between the saturation temperature (for the condenser pressure) and the temperature of the condensate. It is the process whereby the water droplets cling to the tube and lose more heat than the latent heat of condensation. They become subcooled and denser then drop off the tubes into the hotwell.

Condensate depression decreases the overall plant efficiency because more sensible heat must be added to reach saturated conditions in the steam generators. However, the advantage of having a small degree of condensate depression is to ensure adequate net positive suction head (NPSH) for the main condensate pumps and prevent cavitation. Excessive condensate depression causes more air absorbed in condensate and accelerates oxygen corrosion of secondary plant materials.

If condensate depression decreases, then the condensate will be closer to saturation temperature. This produces less NPSH at the suction of the condensate pump, raising the probability of cavitation. However, because the condensate is now at a higher temperature, the steam generators must add less sensible heat to the feedwater to reach saturated conditions, increasing plant efficiency.

At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators. However, probability of cavitation will increase as subcooling (or condensate depression decreases).

Answer/Distractor Analysis:

A. CORRECT. (2/2) See discussion above.

B. WRONG. (1/2) The probability of cavitation will increase as subcooling (or condensate depression decreases).

C. WRONG. (1/2) At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators

D. WRONG. (0/2) At a higher temperature, plant efficiency will increase because less sensible heat must be added in the steam generators. However, probability of cavitation will increase as subcooling (or condensate depression decreases).

75 **16-023** **Points: 1.00**

Given:

- Salem Unit 1 is operating at 80% power, middle of life of the fuel cycle.
- All control rods are withdrawn except D bank is at 190 steps and in manual control.
- Core axial power distribution is peaked below the core midplane.

Which of the following will **INCREASE** the core maximum axial peaking factor? (Assume no operator action is taken unless stated and that main turbine load and core xenon distribution do not change unless stated

A. Turbine load/reactor power is reduced by 10 percent.

B. The control bank D is withdrawn 4 steps.

C. Reactor coolant system boron concentration is reduced by 15 ppm.

D. A fully withdrawn control rod located at the edge of the core drops to the bottom of the core.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that the load reduction will lower hot leg temperatures causing the flux in this portion to increase and resulting in an increase in the axial peaking factor. Incorrect in that the lower hot leg temperature will result in a higher density causing the neutron flux to shift higher in the core due to the increased neutron moderation in these regions (DT gets smaller as power id reduced). This will reduce the axial peaking factor as the flux in the lower potions of the core lowers relative to the average core flux.

B. Incorrect. Plausible because the operator may believe that withdrawing control rods 4 steps will increase the hot leg temperature resulting in a higher flux in this portion and higher peaking factors. Incorrect in that withdrawing one bank of control rods 4 steps will cause the flux to be depressed less (with no change in average flux). This results in a decrease in axial peaking factor.

C. Correct. Reducing boron concentration (diluting) by 15 ppm will insert positive reactivity in the core, resulting on both heat balance (calorimetric) reactor power and average coolant temperature increasing. Also, diluting raises hot leg temperature, making the water in the upper regions of the core less dense (less neutron moderation), thereby shifting the flux to the already flux-dense portion below midplane. Another way to think about this is that the water is colder in the bottom of the core resulting in a greater change in the atoms per cubic centimeter being removed in the bottom of the core. This again adds positive reactivity in the bottom of the core and will shift the flux downward.

D. Incorrect. Plausible because the operator may believe that one control rod dropped in the core will increase the flux in that portion of the core adjacent to the dropped rod resulting in a higher axial peaking factor. Incorrect in that one control rod inserting will affect radial flux peaking, but will have no affect on axial flux peaking because the overall axial flux profile does not shift significantly since it is an axially uniform poison.

Technical References:	INPO GFES Core Thermal Limits (R3)
Proposed References to be provided:	None
Learning Objective:	NOS05CORTHR-02. Refer to INPO GFES Core Thermal Limits Rev. 3. ELO 1.3 Reactor Operation Effects on Peaking Factors.
Question Source:	Bank – INPO GFES P7650
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.14
Level:	RO
Category/System:	16-Thermodynamics / Core Thermal Limits
K/A #:	193009 K1.07 Describe factors that affect peaking and hot channel factors.
Importance:	3.3
Tier/Group	Theory-Tier 4

K/A Match: The K/A is matched because the operator is being asked which plant conditions will affect the axial peaking (or hot channel) factor.

Additional Reference: INPO GFES Core Limits Solution Bank

NATIONAL ACADEMY FOR NUCLEAR TRAINING PWR GFE SOLUTION	THERMODYNAMICS	P7650
	193009 – CORE THERMAL LIMITS	K/A: K1.02 [2.3/2.8]

A reactor is operating at 80 percent power near the middle of a fuel cycle. All control rods are nearly fully withdrawn and in manual control. Core axial power distribution is peaked below the core midplane.

Which one of the following will increase the core maximum axial peaking (or hot channel) factor? (Assume no operator action is taken unless stated, and that main turbine load and core xenon distribution do not change unless stated.)

A. X Turbine load/reactor power is reduced by 10 percent.

B. X The controlling bank of control rods is withdrawn 4 inches.

C. √ Reactor coolant system boron concentration is reduced by 15 ppm.

D. X A fully withdrawn control rod located at the edge of the core drops to the bottom of the core.

Discussion/Calculation:

The steady-state axial peaking (or hot channel) factor is a measure of maximum total power produced in fuel rod, ensuring enthalpy rise in the *entire* core is low enough to preclude reaching DNB. Measurements for this parameter can be obtained using in-core detectors taking a series of measurements across the full length of the core, and determining the *ratio of average flux for one elevation to the average flux for all elevations*. This factor depends on several parameters including fuel loading, control rod bank insertion, and fuel burn-up.

Reducing boron concentration (diluting) by 15 ppm will insert positive reactivity in the core, resulting in both heat balance (calorimetric) reactor power and average coolant temperature increasing. Also, diluting raises hot leg temperature, making the water in the upper regions of the core less dense (less neutron moderation), thereby shifting the flux to the already flux-dense portion below midplane. Another way to think about this is the fact that the water is colder in the bottom of the core resulting in a greater change in the atoms per cubic centimeter being removed in the bottom of the core. This again adds positive reactivity in the bottom of the core and will shift the flux downward. Therefore, Choice "C" is correct.

Answer/Distractor Analysis:

A. WRONG. Reducing reactor power by 10% results in a significant reduction in hot leg temperature with a relatively constant cold leg temperature. As hot leg temperature is reduced, water in the upper portions of the core is at a reduced temperature, resulting in a higher density, causing the neutron flux to shift higher in the core due to increased neutron moderation in these regions (Delta-T gets smaller as power is reduced). This will reduce the axial peaking factor as the flux in the lower portions of the core lowers relative to the average core flux.

B. WRONG. The stem indicates that the core axial distribution is already peaked below the core midplane. Withdrawing one bank of control rods 4 inches will cause the flux to be depressed less (with no change in average flux). This results in an decrease in axial peaking factor.

C. CORRECT. See discussion above.

D. WRONG. One control rod fully inserting will affect radial flux peaking, but will have no affect on axial flux peaking because the overall axial flux profile does not shift significantly since it is an axially uniform poison.

SRO EXAMINATION ANSWER KEY

SALEM 2023 NRC Exam (21-01 ILOT) **– (Rev 0)**

76 **10-144** **Points: 1.00**

Given:

- Salem Unit 2 experienced an automatic reactor trip and safety injection as a result of a partially open PZR PORV
- The affected PZR PORV cannot be isolated

Then:

- The crew transitions to 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant
- RCS pressure is now at 500 psig and slowly LOWERING
- PZR level is rising

Which of the following completes both statements?

1. Five (5) minutes after safety Injection actuated, what is the status of ECCS injection flow?

2. Based on the above conditions, the CRS will direct what procedure NEXT?

A. (1) remains unchanged
(2) 2-EOP-TRIP-3, SI Termination

B. (1) remains unchanged
(2) 2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization

C. (1) rising
(2) 2-EOP-LOCA-2, Post LOCA Cooldown and Depressurization

D. (1) rising
(2) 2-EOP-TRIP-3, SI Termination

Answer: C

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible because the operator may believe that the ECCS injection flow will exceed the break flow thru the PZR PORV and the ECCS flow will quickly stabilize and remain unchanged. Incorrect in that ECCS flow would have risen since SI actuated due to lowering RCS pressure. Second part is incorrect. Plausible because transition to EOP-TRIP-3 from EOP-LOCA-1 is possible, but with RCS pressure still lowering, SI Termination criteria is not met.

B. Incorrect. Plausible because the operator may believe that the ECCS injection flow will exceed the break flow thru the PZR PORV and the ECCS flow will quickly stabilize and remain unchanged. Incorrect in that ECCS flow would have risen since SI actuated due to lowering RCS pressure. Second part is correct.

C. Correct. After the reactor is tripped, PZR level will lower and then eventually rise again due to the failed open PZR PORV and injection from the ECCS Pumps. From the time the reactor tripped and SI actuated, RCS pressure would be lowering resulting in ECCS pump flow(s) rising. Per 2-EOP-LOCA-1, step 12 checks to see if RCS pressure is above 300 psig. If RCS pressure is above 300 psig, then the RCS pressure is above the shutoff head of the RHR pumps and the procedure directs transition to EOP-LOCA-2. If RCS pressure is below the shutoff head of the RHR pumps and RHR flow is indicated, then the operator should stay in EOP-LOCA-1.

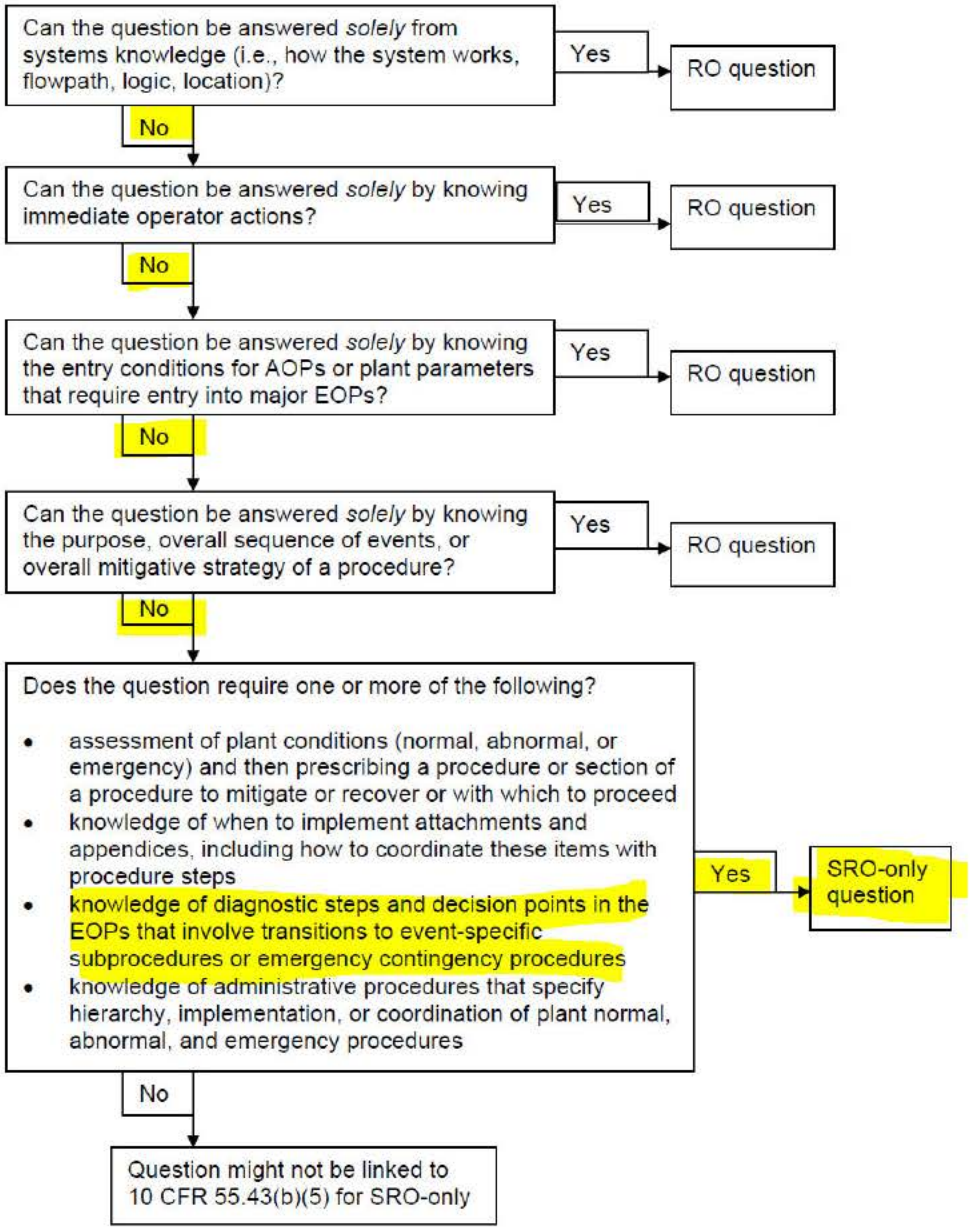
D. Incorrect. First part is correct. Second part is incorrect. Plausible because transition to EOP-TRIP-3 from EOP-LOCA-1 is possible, but with RCS pressure still lowering, SI Termination criteria is not met.

Technical References:	2-EOP-LOCA-2 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05LOCA01-07, Objective 3. For the following analyzed transients/accidents: Inadvertent Depressurization of the RCS - Stuck Open Safety (Condition II - Faults of Moderate Frequency. Describe/State the following: Describe the expected plant response
Question Source:	Modified Bank – Callaway 2021 NRC SRO76
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Pressurizer Vapor Space Accident
K/A #:	APE 8 AA2.15 Ability to determine and/or interpret the following as they apply to the Pressurizer Vapor Space Accident: ECCS status.
Importance:	3.9
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked to determine how ECCS flow injection will behave following SI actuation and RCS pressure due to a vapor space leak and what procedure will the SRO direct based on RCS pressure checks in EOP-LOCA-1.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Parent Question: Callaway 2021 NRC SRO76

Modified Question: Part 1 of the question was changed to address the K/A part on ECCS status. The question is asking the applicant to determine the status of ECCS flow(s) following the SI actuation during a PZR PORV Vapor Space accident and then what procedure will the CRS direct next based on the conditions in the stem.

Question # 76

An automatic reactor trip and safety injection has occurred as a result of lowering RCS pressure.

The crew has just entered E-1, Loss of Reactor or Secondary Coolant and plant conditions are as follows:

	Prior to SI	After SI Occurs
Pzr Pressure	Lowering	500 psig and Lowering
RCS Temperature	Stable	Stable
Pzr Level	Stable	Rising

(1) What accident would result in these conditions?

And

(2) The CRS will direct what procedure NEXT?

A. (1) RCS Cold Leg Leak
(2) ES-1.1, SI Termination

B. (1) RCS Cold Leg Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

C. (1) PZR Vapor Space Leak
(2) ES-1.1, SI Termination

D. (1) PZR Vapor Space Leak
(2) ES-1.2, Post LOCA Cooldown and Depressurization

Answer: D

77 **10-145** **Points: 1.00**

Given:

- A LOCA is in progress on Salem Unit 2.
- The crew is implementing 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant, and is currently at Step 7.4 for assessing conditions for stopping containment spray pumps.
- Containment pressure is 23 psig.



Then:

- A valid PURPLE path exist on Thermal Shock
- The crew enters 2-EOP-FRTS-1, Response to Imminent Pressurized Thermal Shock Conditions
- While performing actions in 2-EOP-FRTS-1, containment pressure decreases to below 13 psig.

In accordance with OP-SA-108-101-2002, EOP Users Guide, which of the following is required and why?

Note: FRP = Functional Restoration Procedure(s)

A. Step 7.4 may be performed if the CRS determines there is no conflict with the procedure in effect.

B. Immediately perform Step 7.4; a Note step in EOP-LOCA-1 specifically states NO FRPs are to be implemented during performance of EOP-LOCA-1.

C. Immediately perform Step 7.4; Step 7.4 is a continuous action step and still in effect.

D. DO NOT perform Step 7.4; EOP-LOCA-1 actions are superseded by the PURPLE path FRP.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the operator may believe that if a conflict arises between actions of an EOP and FRP that the CRS will always have the option to determine the best path to take, for example "Thinking Compliance".

B. Incorrect. Plausible because EOP-LOCA-3 has a Note (N1-1) that does state to NOT implement FRPs while performing actions to establish cold leg recirculation.

C. Incorrect. Plausible because step 7.4 is a continuous action step and would be performed when conditions are met. Incorrect in that continuous actions are not performed when transitions are made to a Red or Purple path FRP.

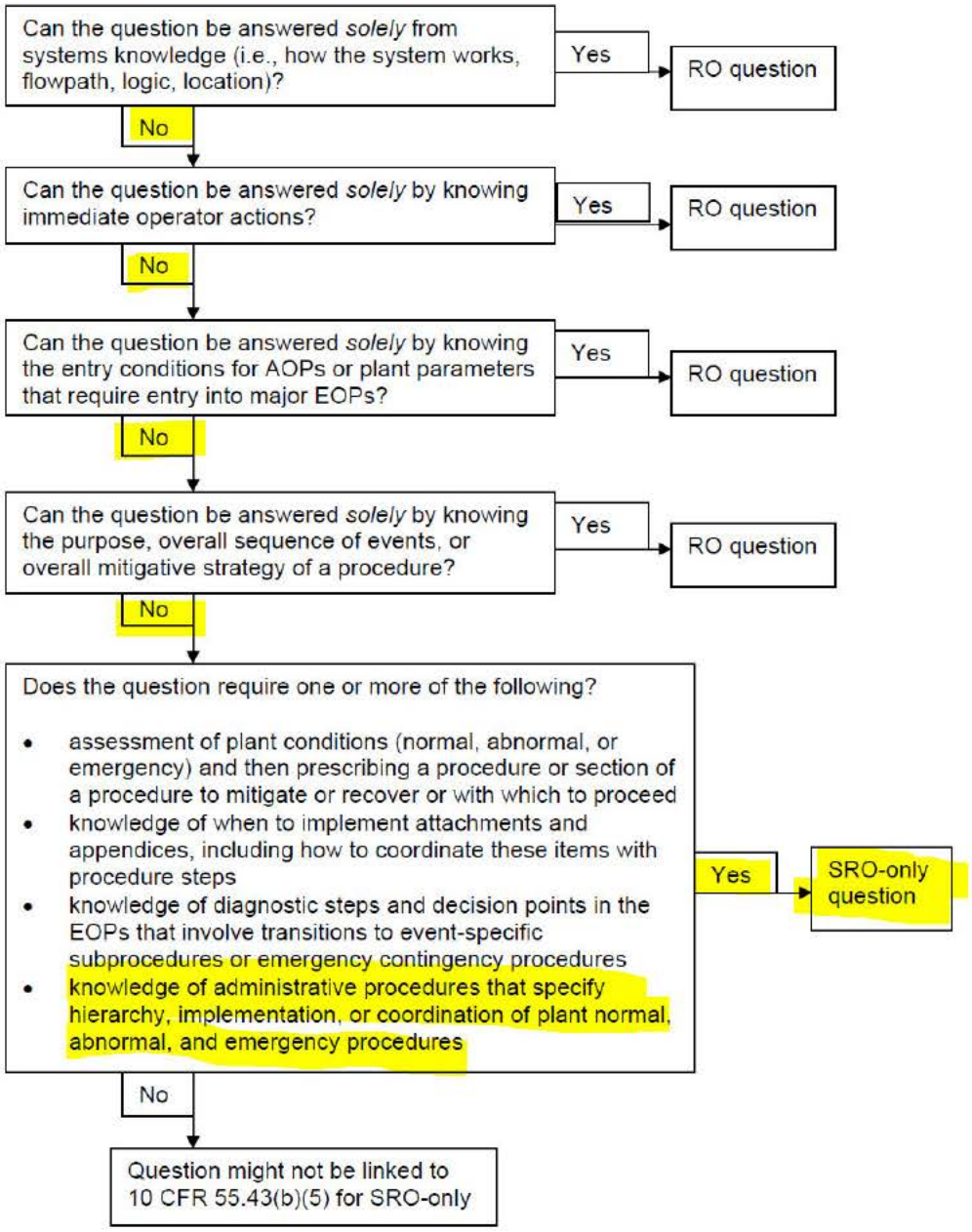
D. Correct. Per OP-SA-108-101-2002 step 2.16.2, a continuous action step should not be performed when a Red or Purple path FRP is in progress. Therefore, even though conditions of LOCA-1 step 7.4 are met, the crew should not stop CS pumps.

Technical References:	OP-SA-108-101-2002 (R0), 2-EOP-CFST-1 (R41)
Proposed References to be provided:	**Embedded step from EOP-LOCA-1**
Learning Objective:	NOS05TRP001-09, Objective 10. Given EOP-CFST-1 and a set of plant conditions, determine whether Functional Restoration Procedures (FRPs) should be implemented and/or terminated, in accordance with OP-SA-108-101-2002 and EOP-CFST-1
Question Source:	Bank - Point Beach 2017 NRC RO74
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.1 / 45.13
Level:	SRO
Category/System:	10-Large Break LOCA / Emergency Procedures/Plan
K/A #:	EPE 11 G2.4.14 Knowledge of general guidelines for emergency and abnormal operating procedures usage
Importance:	4.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked the rules of EOP usage when a Red or Purple path FRP exists and which procedure takes priority when a CAS action condition in the EOP is met.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Additional Reference: OP-SA-108-101-2002

2.16 Continuous Action Steps:

2.16.1. Most guidelines have been provided with a Continuous Action Summary. This is annotated in the upper left corner of each EOP Flowchart page. The particular items of the Continuous Action Summary associated with any guideline are provided in the FOLDOUT PAGE ITEMS document of the Generic Issues Section of the Westinghouse ERG Executive Volume. These foldout page items should be applied to the plant specific EOPs on a per guideline basis. The Continuous Action Summary contains several pieces of information or actions which are normally applicable at any step of the particular guideline. The most important of these actions are guideline transitions which allow immediate response to new symptoms as they appear. The placement of these transitions in the Continuous Action Summary allows prompt response to the appearance of subsequent symptoms.

2.16.2. Certain operator actions provided in a guideline direct or imply continuous performance throughout the remainder of the guideline. A continuous action is an action that is applicable from the point at which it is first encountered until superseded by alternate guidance or stated to be inapplicable. A continuous action generally remains applicable throughout its associated guideline unless otherwise stated, and may apply after a transition is made to another guideline if it does not contain any actions that are inappropriate for the subsequent guideline. For example, a transition contained within a continuous action will not apply after a transition is made to a RED or PURPLE path Functional Restoration Guideline or the LOPA series. Furthermore, actions requiring equipment to be started should not be performed in the LOPA series unless directed in that series. Additionally, Optimal Recovery Guideline actions (EOPs) are not to be performed while a Critical Safety Function is being restored from a RED or PURPLE condition.

78 **11-196** **Points: 1.00**

Given:

- Salem Unit 2 is operating at 30% power
- OHA D-36, RCP VIB HI, is LIT
- 21 RCP parameters are:
  - Motor flange vibration is 3 mils and steady
  - Shaft vibration is 15 mils and steady
- Engineering has determined the vibrations are valid
- The crew enters S2.OP-AB.RCP-0001, RCP Abnormality

What action will the CRS direct?

Note: TS 3.4.1.1, Reactor Coolant Loops and Coolant Circulation
2-EOP-TRIP-1, Reactor Trip or Safety Injection

A. Immediately stop 21 RCP per Attachment 2, Stopping RCPs, of S2.OP-AB.RCP-0001.

B. Determine if a Unit shutdown is required in preparations for removing 21 RCP from service.

C. Stop 21 RCP within 1 hour per TS 3.4.1.1 and place the Unit in MODE 3 within 6 hours.

D. Immediately stop 21 RCP per OHA D-36 Alarm Response Procedure and enter 2-EOP-TRIP-1.

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because motor and flange vibrations are parameters on Attachment 1 of AB.RCP-0001 that may require stopping RCPs per Attachment 2. Incorrect in that the Attachment 1 RCP vibration levels have not been exceeded as of yet.

B. Correct. The RCP vibration levels are below the requirements in Attachment 1 of AB.RCP-1 that require stopping the affected RCP per Attachment 2. The OHA ARP for D-36 directs entry into S2.OP.AB-RCP-0001. Per S2.OP-AB.RCP-0001 Step 3.10 allows the CRS/SM to determine if a Unit shutdown is required.

C. Incorrect. Plausible if the applicant believes that since RCP trip criteria has not been reached per Attachment 1 of AB.RCP-0001, then take the action of TS 3.4.1.1 for a loss of an RCP. Incorrect in that the TS action is to place the unit in MODE 3 within one hour on a loss of RCS flow.

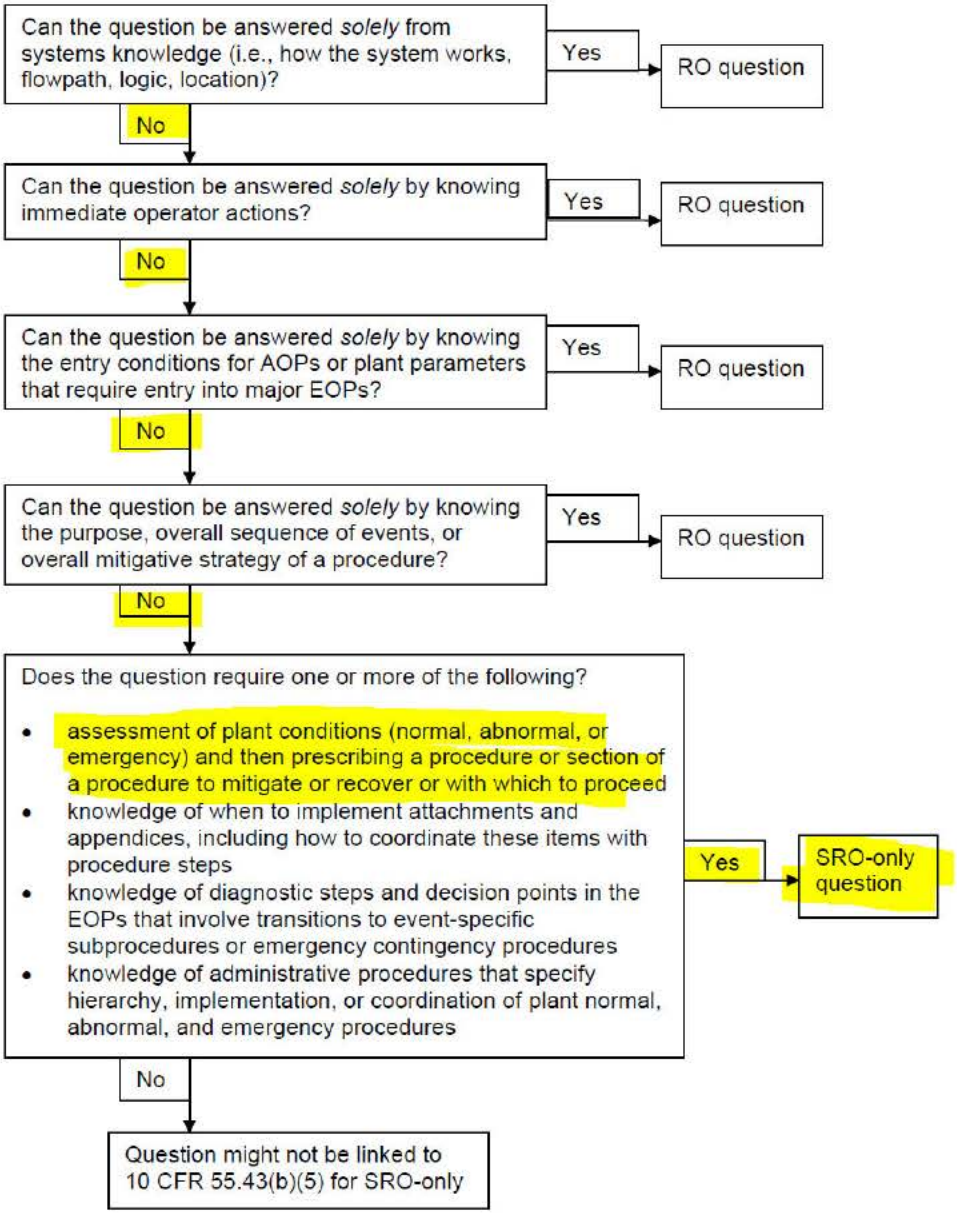
D. Incorrect. Plausible if the applicant believes that the ARP will direct stopping and tripping the reactor which it does not, only directs entry into AB.RCP-0001

Technical References:	S2.OP-AB.RCP-0001 (R28), TS 3.4.1.1.
Proposed References to be provided:	None
Learning Objective:	NOS05ABRCP0-08, Objective 4b. Given a set of initial plant conditions: Describe the final plant condition that is established by the abnormal procedure.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Reactor Coolant Pump Malfunctions
K/A #:	APE 15 AA2.16 Ability to determine and/or interpret the following as they apply to RCP Malfunctions: RCP vibration
Importance:	3.5
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked to interpret RCP vibrations and determine the required course of action per S2.OP-AB.RCP-0001.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Additional Reference: S2.OP-AB.RCP-0001

ATTACHMENT 1
(Page 1 of 2)

CONTINUOUS ACTION SUMMARY

___ 1.0 IF AT ANY TIME, any of the following validated RCP conditions exist, THEN GO TO Attachment 2, Stopping Reactor Coolant Pumps.

_____ Time

- Component Cooling Water flow is lost to ALL RCPs.
- Component Cooling Water flow not restored within 5 minutes of initial loss:
  - OHA-D-20, 21 RCP BRG CLG WTR FLO LO
  - OHA-D-21, 22 RCP BRG CLG WTR FLO LO
  - OHA-D-22, 23 RCP BRG CLG WTR FLO LO
  - OHA-D-23, 24 RCP BRG CLG WTR FLO LO
- RCP Seal Injection Flow AND RCP Thermal Barrier Component Cooling flows are lost concurrently (RCPs should be secured within 2 minutes to prevent RCP damage).
- Shaft vibration greater than 20 mils or motor flange vibration greater than 5 mils [C0379]
- Motor Bearing temperature greater than 175°F
- Motor Winding temperature greater than 302°F, (Modes 1-5)
  (Also, **REFER TO** Attachment 3 (page 2 of 2) for Alternate Indications to validate indicated rise in RCP Motor Winding Temperature)
- Seal Water Outlet temperature greater than 190°F
- #1 Seal Water Leakoff flow less than 0.8 gpm and reverses to greater than or equal to 6.0 gpm or pump bearing/seal inlet temperature rise
- #1 Seal Water Leakoff flow less than 0.8 gpm with rising pump bearing/seal inlet temperatures
- #1 Seal Water Leakoff flow greater than or equal to 6 gpm

Additional Reference:

___ 3.9 IF the RCP malfunction is NOT determined,
THEN **RETURN** to Step 3.1.

NOTE
___ Following a Loss of Seal Injection event, with RCPs having a seal leakoff less than 2.5 gpm, RCS cooldown will help to lower seal leakoff temperatures.

___ 3.10 IF Unit shutdown is determined by CRS/SM,
THEN:

___ A. **INITIATE** Unit shutdown IAW applicable Integrated Operating Procedures.

___ B. **NOTIFY** CRS/SM to refer to the following:

- Technical Specifications
- Event Classification Guide

G. Step 3.10, provides direction to perform a controlled Unit shutdown when unacceptable RCP conditions exist. The unit is shutdown in a period of 8 hours or less and the affected RCP(s) are removed from service allowing a more detailed inspection and analysis. Notification for CRS/SM to evaluate plant conditions. Plant Technical Specifications and the Event Classification Guide are evaluated for applicability.

79 **14-081** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 5
- 21 RHR Pump is in shutdown cooling
- 22 RHR Pump is in standby

Then:

- 21 RHR Pump trips
- The crew enters S2.OP-AB.RHR-0001, Loss of RHR
- Attempt to start 22 RHR Pump results in the breaker tripping free
- STA reports Time to Core Boiling is less than 15 minutes
- Field operator reports no protective relay flags are present and no obvious reason for the 4KV breaker for 22 RHR Pump to trip free, and an acrid smell is observed near the 21 RHR 4KV breaker cubicle

In accordance with OP-SA-108-106-1001, Equipment Control and Large Motor Starting Criteria, complete the following statement concerning breaker re-closure for 22 RHR Pump?

Based on the above conditions, the __(1)__ can authorize one attempt at reclosing the breaker without an investigation ONLY during __(2)__.

A. (1) Shift Manager
(2) emergency conditions

B. (1) Plant Manager
(2) emergency conditions

C. (1) Shift Manager
(2) emergency conditions AND with Maintenance Director concurrence

D. (1) Plant Manager
(2) emergency conditions AND with Maintenance Director concurrence

Answer: A

Explanation / Justification:

A. Correct. Per OP-SA-108-106-1001 step 4.2.2, if an emergency condition exists, as determined by the SM/CRS, and a breaker fails to close, one attempt at reclosure may be made without conducting an investigation, if that piece of equipment is essential for maintaining plant stability..

B. Incorrect. Plausible because the conditions in the stem are severe and the SRO may believe that the Plant Manager level of management is required to accept responsibility if the RHR motor is damaged.

C. Incorrect. Plausible because the SRO may think that before reclosing the breaker to a important safety equipment that Maintenance should be involved in the decision process.

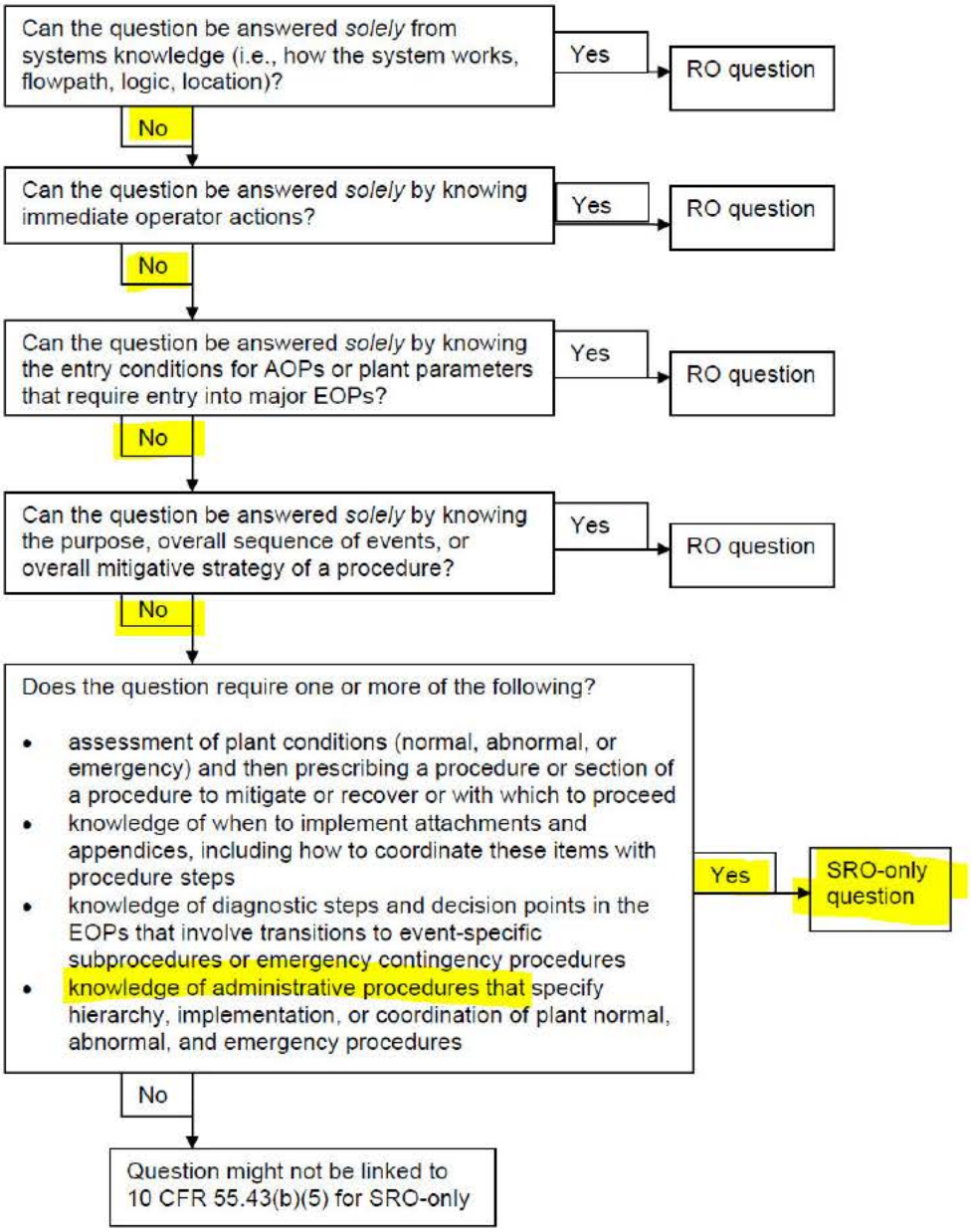
D. Incorrect. Plausible because the SRO may think that before reclosing the breaker to a important safety equipment that Maintenance should be involved in the decision process.

Technical References:	OP-SA-108-106-1001 (R1)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-15, Objective 5. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative Procedures
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.10 / 45.13
Level:	SRO
Category/System:	14-Loss of RHR System / Conduct of Operations
K/A #:	APE 25 G2.1.1 Knowledge of conduct of operations requirements
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being ask knowledge of the breaker reclosure policy when the breaker trips free while attempting to start an RHR pump during loss of RHR cooling.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



80 **10-146** **Points: 1.00**

Given:

- Salem Unit 2 reactor failed to automatically trip following a valid RPS trip signal
- All attempts to trip the reactor from the main control room have been unsuccessful
- The crew enters 2-EOP-FRSM-1, Response to Nuclear Power Generation, and all immediate actions are complete

In accordance with 2-EOP-FRSM-1, which of the following describes how the turbine trip status is checked AND what subsequent action will the CRS direct if the turbine trip status check is NOT met following immediate actions of 2-EOP-FRSM-1?

A. ALL turbine stop valve closed bi-stables LIT on 2RP4; dispatch an operator to locally trip the turbine at the front standard.

B. ALL turbine stop valve closed bi-stables LIT on 2RP4; direct the control room operator to stop all EHC pumps.

C. 2 out 3 Auto Stop Oil low pressure bi-stables LIT on 2RP4; direct the control room operator to stop all EHC pumps.

D. 2 out 3 Auto Stop Oil low pressure bi-stables LIT on 2RP4; dispatch an operator to locally trip the turbine at the front standard.

Answer: A

Explanation / Justification:

A. Correct. Per 2-EOP-FRSM-1 Step 6, the turbine is tripped when all stop valves are closed. This can be verified by observing the bi-stables on 2RP4. If the turbine fails to trip at Step 6, then the CRS will dispatch an operator to the turbine deck to locally trip the turbine at the front standard.

B. Incorrect. First part is correct. Second part is incorrect. Plausible because stopping EHC pumps is possible from the control room. Incorrect in that FRSM-1 step 6 directs locally tripping the turbine at the front standard.

C. Incorrect. First part is incorrect. Plausible because 2 out of 3 ASO bi-stables tripped is a demand for a turbine trip. Incorrect in that ASO indications is not a direct indication of a turbine trip. Second part is incorrect. Plausible because stopping EHC pumps is possible from the control room. Incorrect in that FRSM-1 step 6 directs locally tripping the turbine at the front standard.

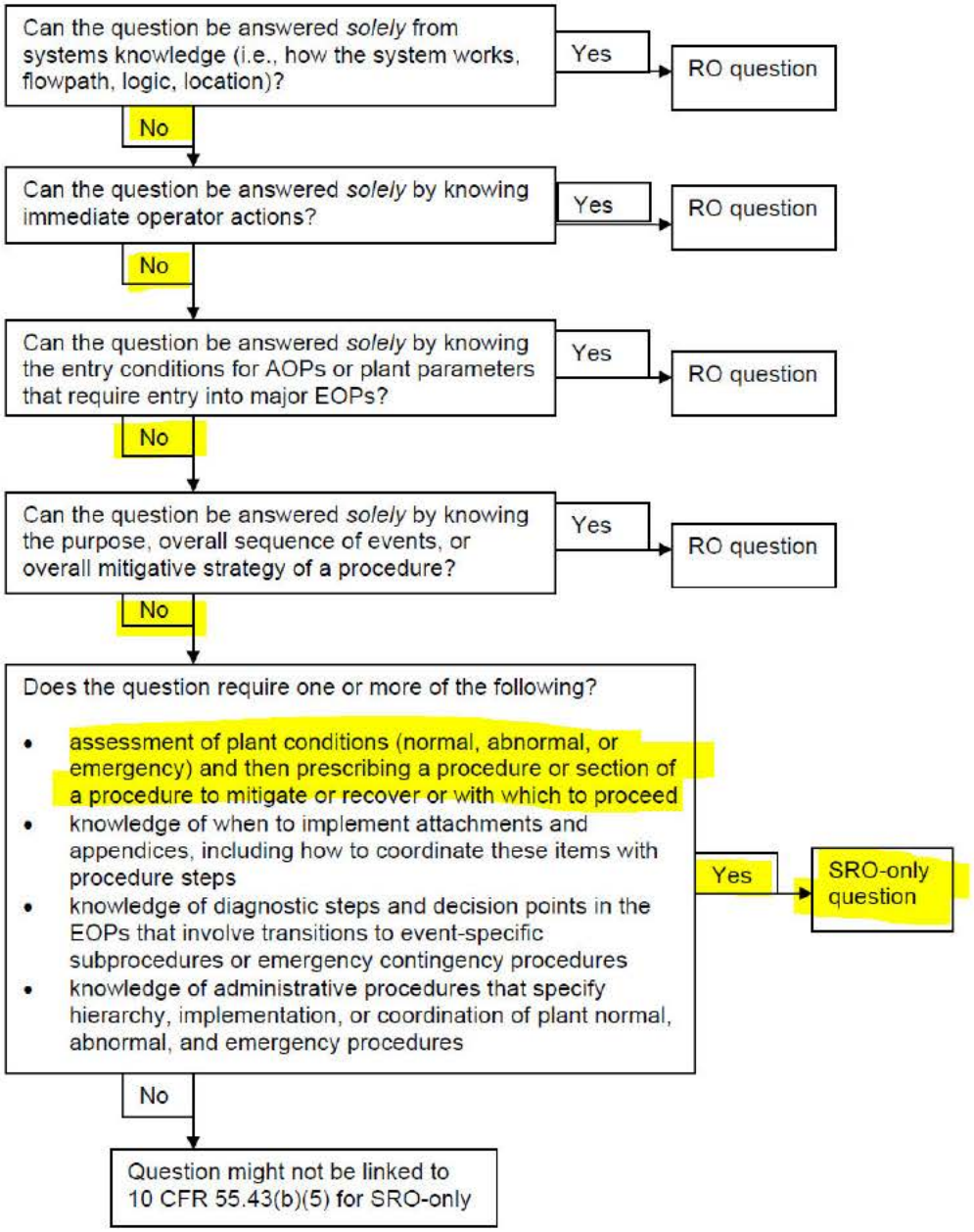
D. Incorrect. First part is incorrect. Plausible because 2 out of 3 ASO bi-stables tripped is a demand for a turbine trip. Incorrect in that ASO indications is not a direct indication of a turbine trip. Second part is correct.

Technical References:	2-EOP-FRSM-1 (R43)
Proposed References to be provided:	None
Learning Objective:	NOS05FRSM00-05, Objective 4.iii. Describe the plant response to actions taken in the following EOP step sequence(s): Turbine Trip Verification.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Anticipated Transient Without Scram
K/A #:	EPE 29 EA2.09 Ability to determine and/or interpret the following as they apply to ATWS: Occurrence of a main turbine trip.
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the candidate is being asked how to determine if the main turbine is tripped per EOP-FRSM-1 and what action is required if the turbine fails to trip.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



81 **11-197** **Points: 1.00**

Given:

- Salem Unit 2 is at 75% power with a power ascension in progress
- Thunderstorm Alert reported for the site
- 21 and 22 Condensate Pumps are running
- All Heater Drain Pumps are running

Then:

- Lightning strike in the switchyard causes 22 Condensate Pump and ALL Heater Drain Pumps to trip
- The crew enters S2.OP-AB.CN-0001, Main Feedwater/Condensate System Abnormality

Based on the above conditions, which of the following completes both statements?

1. What is the maximum power level allowed?
2. What procedure will the CRS direct, if the load reduction ramp rate is 2% per minute?

[REFERENCE PROVIDED]

A. (1) 65%
(2) S2.OP-IO.ZZ-0004, Power Operation

B. (1) 65%
(2) S2.OP-AB.LOAD-0001, Rapid Load Reduction

C. (1) 30%
(2) S2.OP-AB.LOAD-0001, Rapid Load Reduction

D. (1) 30%
(2) S2.OP-IO.ZZ-0004, Power Operation

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the SRO may incorrectly interpret the table in Attachment 3 and may incorrectly interpret the ramp rate that requires use of S2.OP-AB.LOAD-0001.

B. Incorrect. Plausible because the SRO may incorrectly interpret the table in Attachment 3.

C. Correct. Per S2.OP-AB.CN-0001 Attachment 2 Section 4.0, with only 1 Condensate Pump and no Heater Drain Pumps running, the maximum power level allowed is 30% power. Per S2.OP-IO.ZZ-0004 step 2.2.14, ramp rates greater than 1% per minute should be performed using S2.OP-AB.LOAD-0001.

D. Incorrect. Plausible because the SRO may incorrectly interpret the ramp rate that requires use of S2.OP-AB.CN-0001.

Technical References:	S2.OP-AB.CN-0001 (R32), S2.OP-IO.ZZ-0004 (R87)
Proposed References to be provided:	**Attachment 3 from S2.OP-AB.CN-0001**
Learning Objective:	NOS05ABCN01-07, Objective 4.b, Given a set of initial plant conditions: Describe the plant response to actions taken in the abnormal procedure
Question Source:	Bank – Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.12
Level:	SRO
Category/System:	11-Loss of Main Feedwater / Conduct of Operations
K/A #:	APE 54 G2.1.25 Ability to interpret reference materials, such as graphs, curves, and tables (reference potential).
Importance:	4.2
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 1

K/A Match: The K/A is matched because the SRO is required to assess plant conditions and determine which procedure to implement for the load reduction. In addition, the questions asks the SRO to determine the power level using the table in S2.OP-AB.CN-0001.

Additional Reference:

S2.OP-AB.CN-0001(Q)

ATTACHMENT 2, Programs
(Page 3 of 3)

4.0 MAXIMUM RX POWER LEVEL WITH EQUIPMENT OUT OF SERVICE

Equipment in Service		Maximum Reactor % Power	Reduction Rate
Condensate Pumps	Heater Drain Pumps		
3	2	100	≤ 5%/minute on loss of CN or HD Pump(s)
3	1	90	
3	0	85	
2	3	85	
2	2	80	
2	1	75	
2	0	65	
1	0	30	

5.0 LOAD LIMITATIONS WITH FEEDWATER HEATER(S) OUT OF SERVICE

Combinations of Feed Water Heaters Out of Service	Load Limitations Mw
1, 2, or 3 of the 26 Heaters	1224
1 Group of 21 & 22 Heaters	1098
1 Group of 23, 24 & 25 Heaters	1142
2 Groups of 23, 24 & 25 Heaters	1060
1 Group of 23, 24 & 25 Heaters & a 26 Heater	1018
3 Groups of 23, 24 & 25 Heaters	979
1 Train of 21, 22, 23, 24, 25 & 26 Heaters	893

Additional Reference:

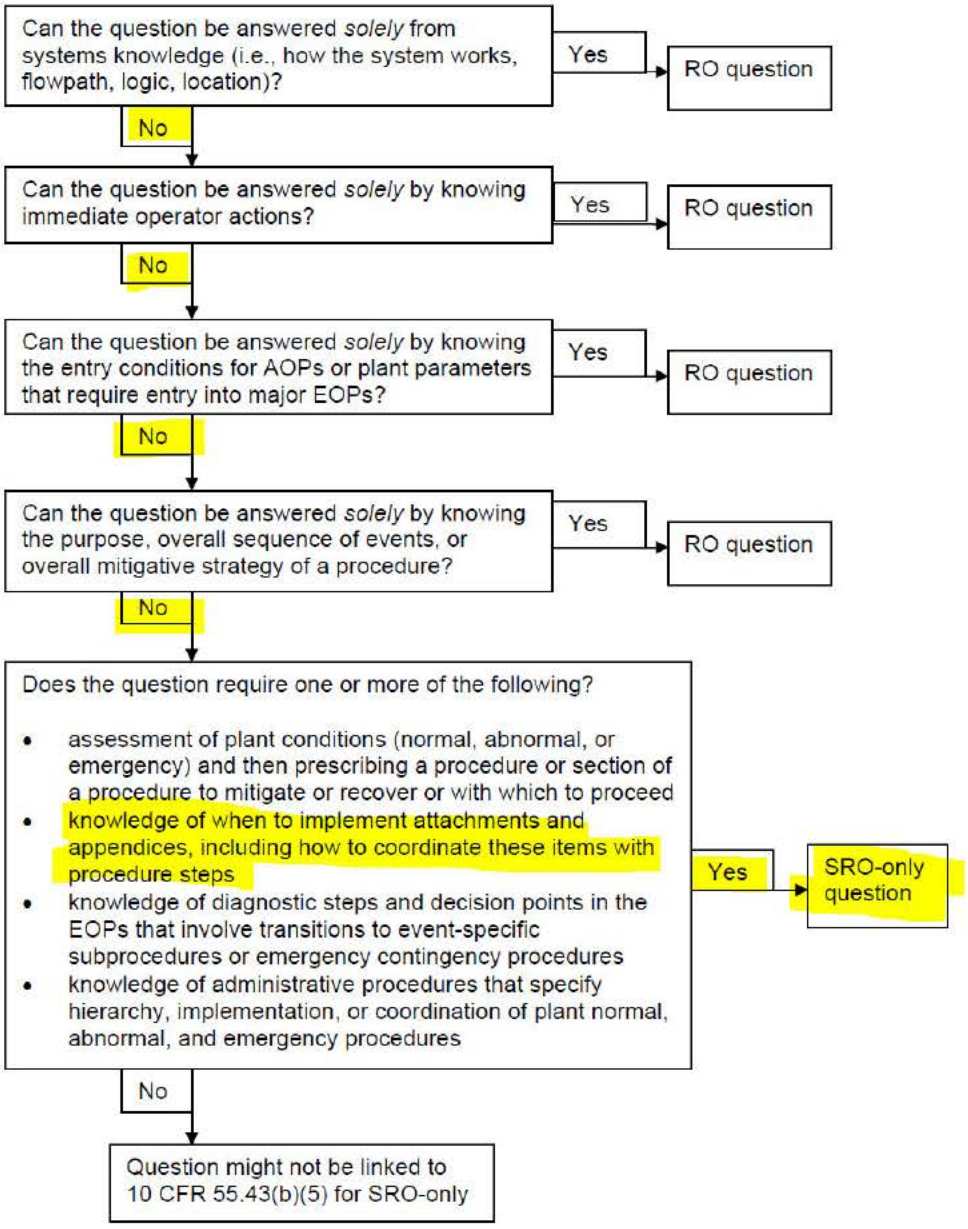
_____ 2.2.13. Based on the response of the feed water regulating valves during the shutdown for S2R23 it is acceptable to trip the reactor at less than or equal to 35% Reactor Power as part of the normal shutdown
sequence. This does NOT prevent the operating crew from taking the necessary actions at any power level to place the plant in a safe
condition if plant conditions dictate.

_____ 2.2.14. Power reduction ramp rates greater than 1%/min should be performed IAW S2 OP AB LOAD 0001(Q), Rapid Load Reduction.

_____ 2.2.15. Implement Attachment 7, Controlling Feed Regulating Valve Vibration, when peak vibration levels on any BF19 Feed regulator valve exceeds 2 inches per second in any direction to prevent exceeding the positioner feedback arm design limits of 10 in/sec in the vertical and axial directions and 5 in/sec in the lateral direction.]

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



82 **11-198** **Points: 1.00**

Given:

- Salem Unit 2 is performing a reactor startup
- The P-6 bistable light is now LIT on 2RP4
- Source Range NIS has been blocked

Then:

- The following indications are observed on the front panel of the 2N35 drawer;
  - INSTRUMENT POWER ON lamp is NOT LIT
  - CONTROL POWER ON lamp is LIT

Which of the following completes both statements?

1. When the P-6 light is lit, the Intermediate Range NIS channels should indicate __(1)___.

2. Based on the above conditions, the CRS will direct actions per ___(2)___.

A. (1) 1E-4% power
(2) 2-EOP-TRIP-1, Reactor Trip or Safety Injection

B. (1) 1E-5% power
(2) 2-EOP-TRIP-1, Reactor Trip or Safety Injection

C. (1) 1E-5% power
(2). S2.OP-AB.NIS-0001, NIS Malfunctions

D. (1) 1E-4% power
(2) S2.OP-AB.NIS-0001, NIS Malfunctions

Answer: C

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse this indication with the indication for when a SR reactor trip will occur when the IR NIS reads 1E-4%. Part 2 is incorrect. Plausible because the operator may believe that a loss of instrument power will cause the RPS bistables to de-energize causing a reactor trip.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because the operator may believe that a loss of instrument power will cause the RPS bistables to de-energize causing a reactor trip.

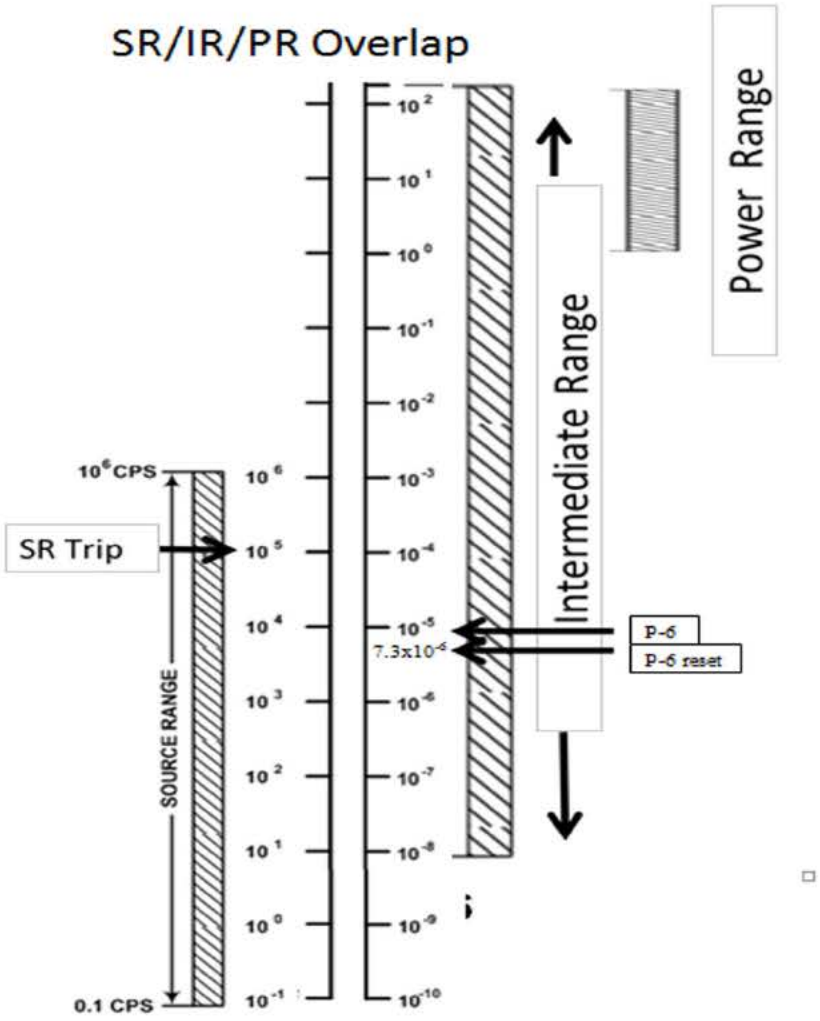
C. Correct. When P-6 is LIT, the IR NIS would indicate 1E-5% power. When a loss of instrument power occurs, this will effect power to the detector and result the channel reading zero. A reactor trip will not occur on loss of instrument power in the drawer because the Control Power supplies the bistables to RPS. The CRS should direct entry into S2.OP-AB.NIS-0001.

D. Incorrect. Part 1 is incorrect. Plausible because the operator can confuse this indication with the indication for when a SR reactor trip will occur when the IR NIS reads 1E-4%. Part 2 is correct.

Technical References:	NOS05ABNIS-04
Proposed References to be provided:	None
Learning Objective:	NOS05ABNIS-04, Objective 1. Describe the operation of the following as applied to S2.OP-AB.NIS-0001(Q), in accordance with this lesson plan: Intermediate Range Instrumentation
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Loss of Intermediate Range Nuclear Instrumentation
K/A #:	APE 33 AA2.01 Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Equivalency among source range, intermediate range, and power range channel readings.
Importance:	3.4
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

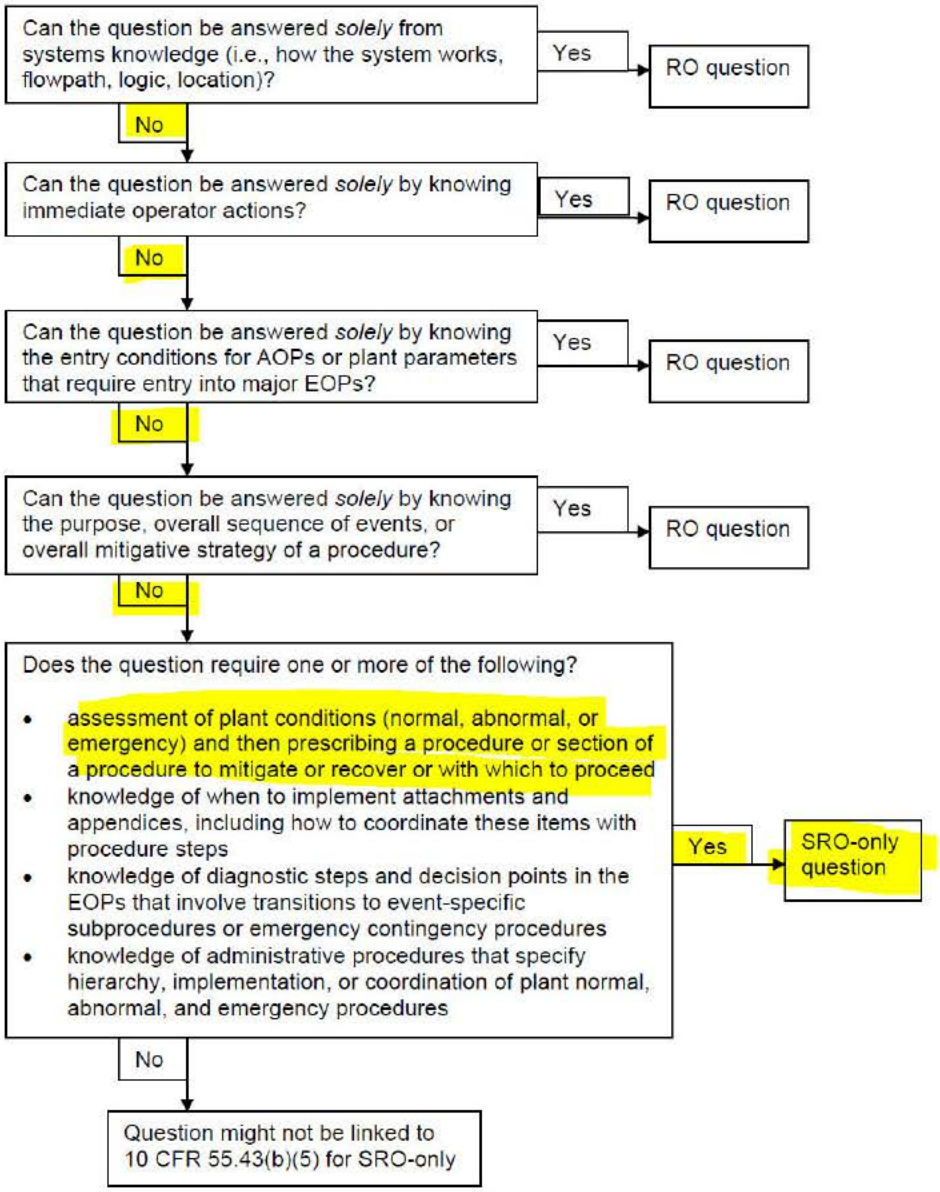
K/A Match: The K/A is matched because the candidate is being asked what the equivalency reading would be between the SR and IR NIS when the P-6 bistable is LIT and based on the loss of instrument power to 2N35 IR NIS channel what procedure will the CRS direct based on plant impact.

Additional Reference:



SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



83 **11-199** **Points: 1.00**

Given:

- BOTH Salem Control Rooms have been evacuated due to toxic gas

Per S2.OP-AB.CR-0001, Control Room Evacuation, which of the following completes both statements?

1. EOPs __(1)__ applicable during Control Room Evacuation.
2. The "Caution" statement in Attachment 3 alerts the CRS to monitor SG pressures at the Hot Shutdown Panel to avoid __(2)__.

A. (1) are NOT
(2) exceeding Tech Spec RCS cooldown rate of 100 °F per hour

B. (1) are NOT
(2) Safety Injection on steam line differential pressure

C. (1) are
(2) exceeding Tech Spec RCS cooldown rate of 100 °F per hour

D. (1) are
(2) Safety Injection on steam line differential pressure

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SRO may that TS limit of 100 ° F per hour still apply in this procedure. Incorrect in that the RCS cooldown limit is 25 °F per hour IAW Attachment 3 step 24.0.

B. Correct. Per the Note in section 2.0 Immediate Actions in S2.OP-AB.CR-0001, EOPs are NOT applicable during Control Room Evacuation. EOPs should be used for information only or as directed by the TSC while performing this procedure. The CRS will monitor SG pressures at the Hot Shutdown Panel IAW Attachment 3 step 5.0. A Caution statement in step 5.0 cautions the CRS that a steam line differential pressure may occur if DP exceeds 100 psi differential pressure

C. Incorrect. Plausible because EOP-TRIP-1 is the typical procedure to enter following a reactor trip. Plausible because the SRO may that TS limit of 100 ° F per hour still apply in this procedure. Incorrect in that the RCS cooldown limit is 25 °F per hour IAW Attachment 3 step 24.0.

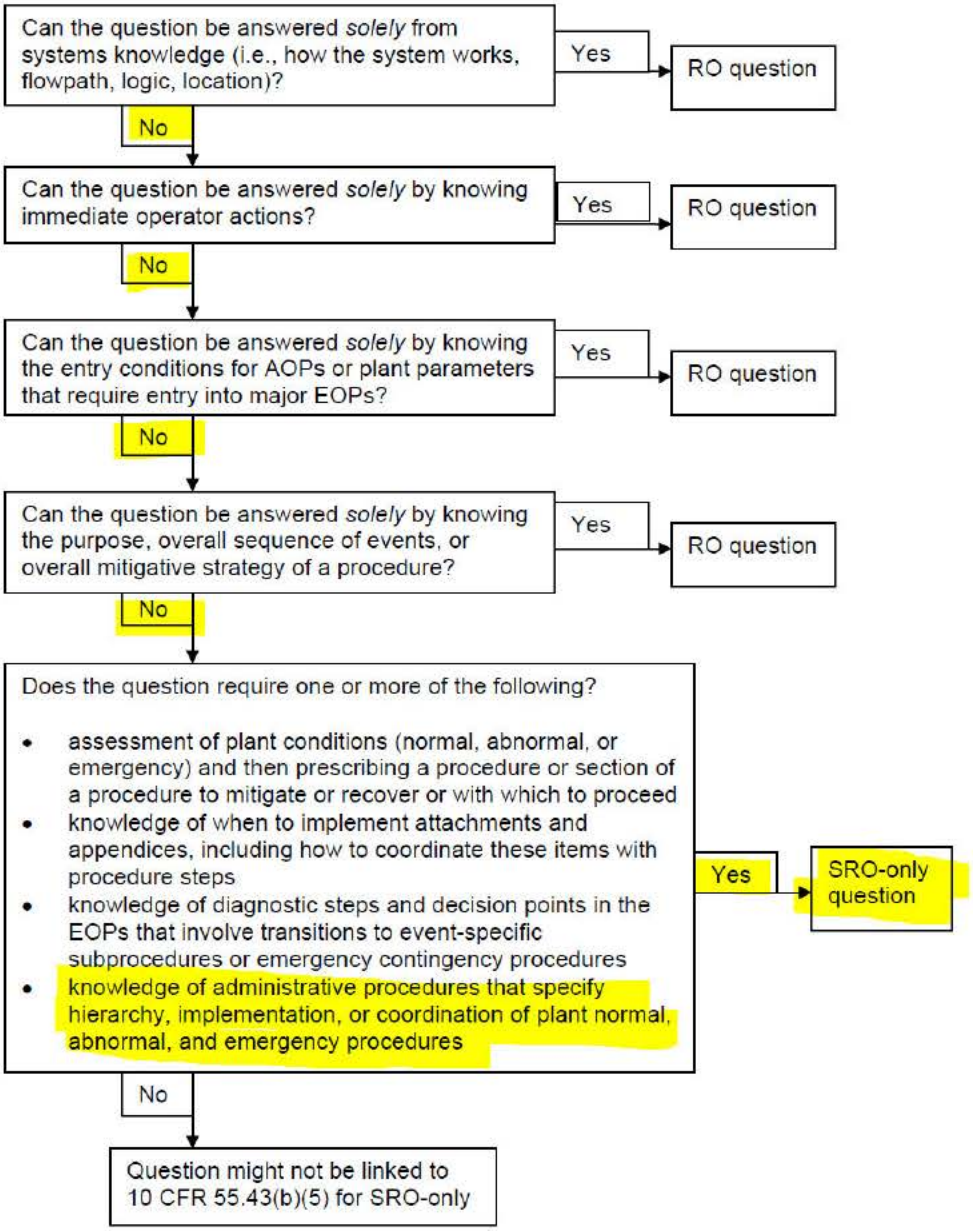
D. Incorrect. Plausible because EOP-TRIP-1 is the typical procedure to enter following a reactor trip.

Technical References:	S2.OP-AB.CR-0001 (R24)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCR01-06 Objective 2. Describe, in general terms, the actions taken in S1/S2.OP-AB.CR-0001 and the bases for the actions.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	11-Control Room Evacuation / Emergency Procedures/Plan
K/A #:	APE 68 G2.4.20 Knowledge of the operational implications of emergency and abnormal operating procedures warnings, cautions, and notes.
Importance:	4.6
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the question is asking the SRO knowledge of a note in the Control Room Evacuation abnormal procedure and how to apply the note.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



84 **10-147** **Points: 1.00**

Given:

- Salem Unit 2 is performing 2-EOP-FRCC-1, Response to Inadequate Core Cooling
- All steam generators are intact and at approximately 1040 psig
- MSIVs are open
- The crew is preparing to depressurize the steam generators in order to inject accumulators

Which of the following completes both statements?

1. The operators will depressurize the steam generators using steam dumps at a __(1)__.
2. If after all available RCPs are started, all PORVs and head vents are open, and CETs are greater than 1200 °F and rising, the CRS will direct transition to __(2)__.

A. (1) maximum rate while attempting to avoid a Main Steam Isolation
(2) SC.OP-AM.SAG-0002, Severe Accident Control Room Guidelines After TSC is Functional

B. (1) maximum rate while attempting to avoid a Main Steam Isolation
(2) SC.OP-AM.SAG-0001, Initial Response

C. (1) rate not to exceed 100 °F per hour
(2) SC.OP-AM.SAG-0002, Severe Accident Control Room Guidelines After TSC is Functional

D. (1) rate not to exceed 100 °F per hour
(2) SC.OP-AM.SAG-0001, Initial Response

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because SC.OP-AM.SAG-0002 is a procedure used but only when directed from SC.OP-AM.SAG-0001 and the TSC is staffed.

B. Correct. Per 2-EOP-FRCC-1 step 11, the crew will depressurize the intact SGs by using the steam dumps or MS10s at a maximum rate. If all attempts to restore core cooling has failed and CETs are greater than 1200 °F and rising and all available RCPs are running, then step 20 directs transition to SC.OP-AM.SAG-0001.

C. Incorrect. Part 1 is incorrect. Plausible because this RCS cooldown rate is used on other EOPs to avoid violating tech spec limits. Part 2 is incorrect. Plausible because SC.OP-AM.SAG-0002 is a procedure used but only when directed from SC.OP-AM.SAG-0001 and the TSC is staffed.

D. Incorrect. Part 1 is incorrect. Plausible because this RCS cooldown rate is used on other EOPs to avoid violating tech spec limits. Part 2 is correct.

Technical References:	2-EOP-FRCC-1 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05FRCC00-06 Objective 2.a. Describe the EOP mitigation strategy for the following: Response to Inadequate Core Cooling
Question Source:	Modified Bank – Diablo Canyon 2019 NRC SRO84
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Inadequate Core Cooling
K/A #:	EPE 74 EA2.12 Ability to determine and/or interpret the following as they apply to the Inadequate Core Cooling: RCS cooldown rate
Importance:	3.8
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the SRO must determine the RCS cooldown rate during depressurizing SGs to inject the accumulators and select the procedure to enter if all attempts to restore core cooling fails in order to mitigate inadequate core cooling.

Parent Question: Diablo Canyon 2019 NRC SRO84

Modified Justification: Part 1 of this questions was changed to test the SRO on the cooldown rate in 2-EOP-FRCC-1.

Question 84

GIVEN

- The crew is performing EOP FR-C.1, Response to Inadequate Core Cooling
- All RCPs are available
- Containment pressure is 1.1 psig and stable
- Narrow Range Steam Generator Levels are:
  - 1-1 ~ 24%
  - 1-2 ~ 12%
  - 1-3 ~ 18%
  - 1-4 ~ 0%

1) In accordance with step 19, Restore Temporary Core Cooling, of EOP FR-C.1, what is the MAXIMUM number of RCS cooling loops available?

2) If, after the RCPs are started in the available RCS cooling loops and all PORVs and head vents are open, core exit thermocouples are greater than 1200°F and rising, the Shift Foreman should:

A. 1) two
   2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

B. 1) two
   2) return to step 1, Ensure ECCS Valve Alignment – Proper Emergency Alignment

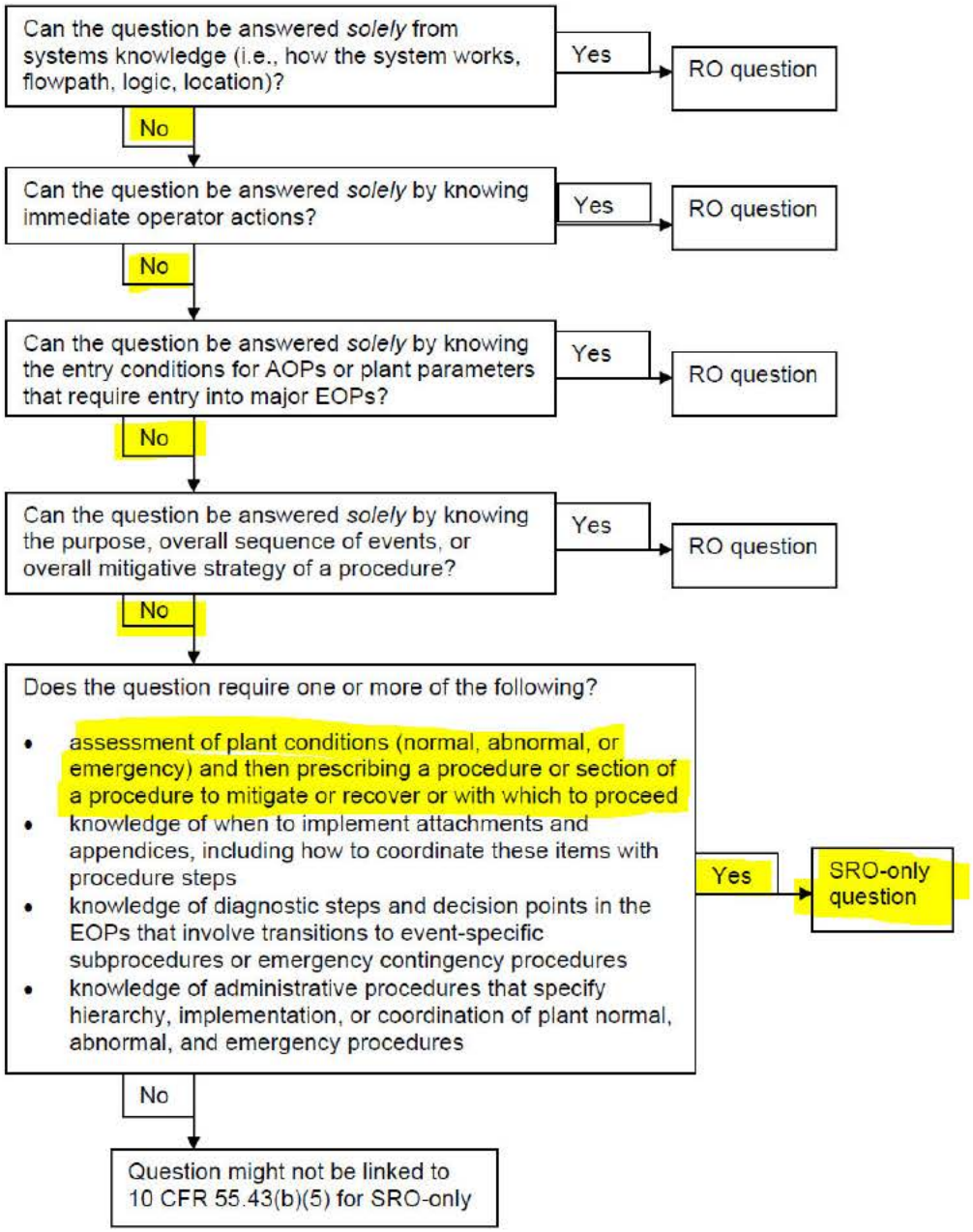
C. 1) three
   2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

D. 1) three
   2) return to step 1, Ensure ECCS Valve Alignment – Proper Emergency Alignment

Proposed Answer: A. 1) two
2) GO TO SACRG-1, Severe Accident Control Room Guideline Initial Response

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



85 **10-148** **Points: 1.00**

Given:

Redacted due to security-related information

Redacted due to security-related information

Redacted due to security-related information

Answer: A

Explanation / Justification:

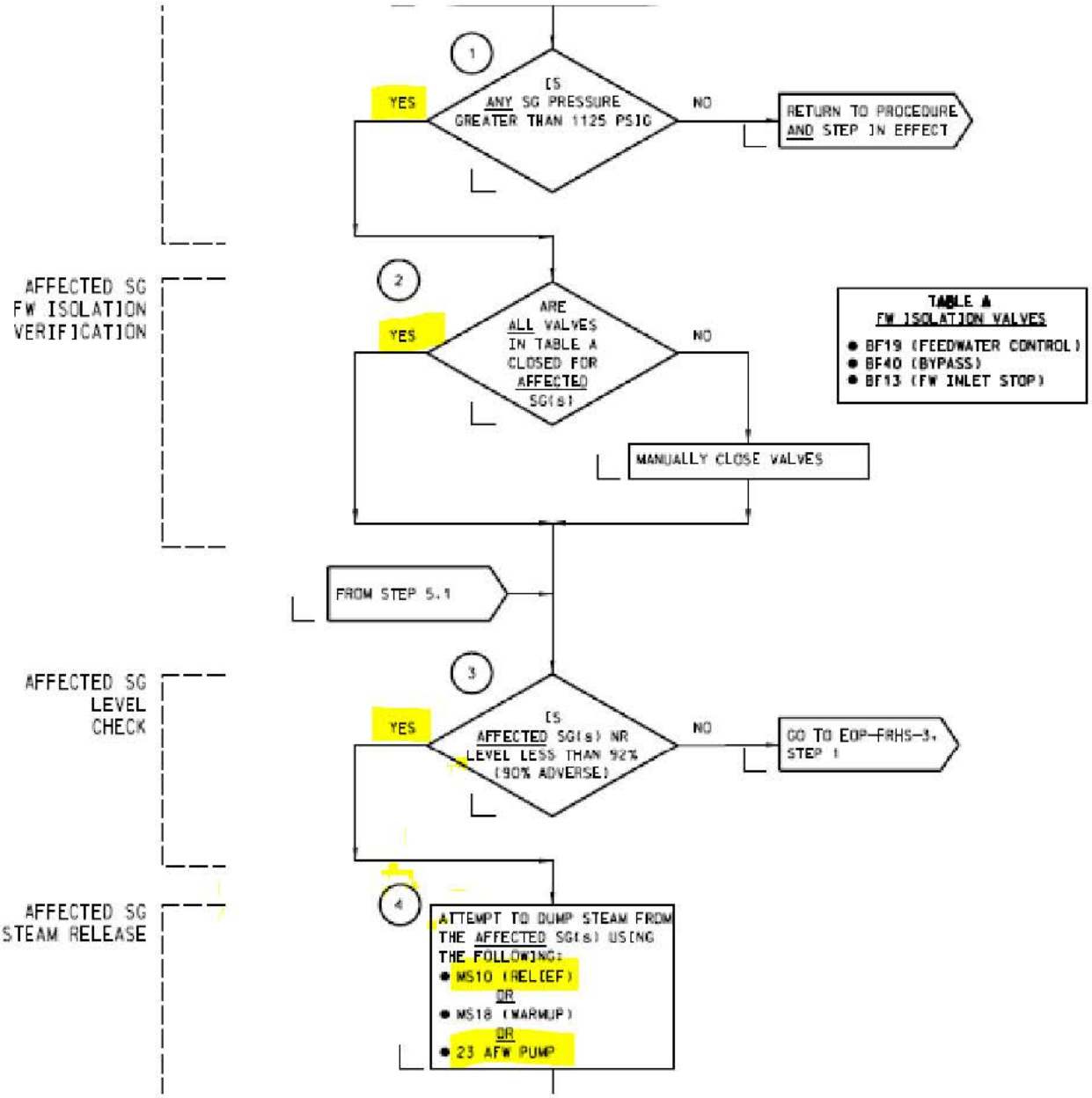
A. [redacted]

Technical References:	2-EOP-FRHS-2 (R41), [redacted]
Proposed References to be provided:	None
Learning Objective:	NOS05ABCR04-03, Objective 1. Describe, in general terms, the actions taken in [redacted] and the bases for the actions .
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	10-Steam Generator Overpressure / Emergency Procedures/Plan
K/A #:	W E13 G2.4.28 Knowledge of the procedures relating to a security event (ensure that the test item includes no safeguards information).
Importance:	4.1
Tier/Group	Emergency and Abnormal Plant Evolutions-Tier 1/Group 2

K/A Match: The K/A is matched because the SRO is being asked knowledge of site security procedures and how to implement them during a SG overpressure event.

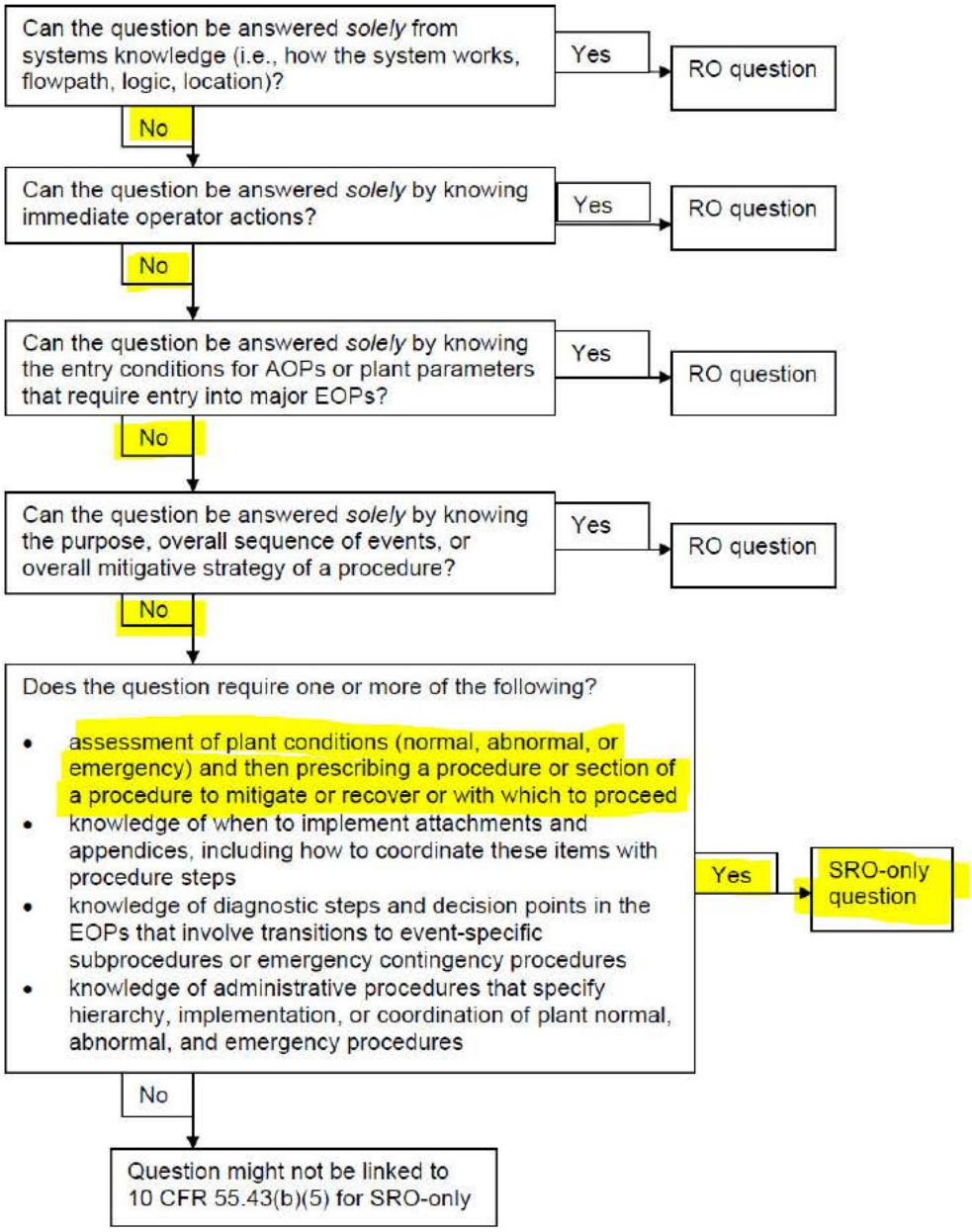
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Additional References: 2-EOP-FRHS-2



SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



86 **04-071** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- Console Alarm, STANDPIPE LEVEL-LO, is now LIT
- The RO reports the following for 21 RCP:
  - Seal Leak-off recorder indicates 1.5 gpm and steady
  - Seal Injection Flow indicates 8.5 gpm
  - Seal Inlet and Outlet Temperatures are indicating 109 °F and steady

Which of the following completes both statements?

1. Which RCP seal failed?
2. What action(s) will the CRS direct the crew?

Note: ARP = Alarm Response Procedure

S2.OP-AB.RCP-0001, Reactor Coolant Pump Abnormality

A. 1) #3 seal
2) Perform an orderly shutdown and stop 21 RCP within 8 hours per S2.OP-AB.RCP-0001.

B. 1) #3 seal
2) Make-up to the standpipe per the ARP; power operation may continue.

C. 1) #2 seal
2) Make-up to the standpipe per the ARP; power operation may continue.

D. 1) #2 seal
2) Perform an orderly shutdown and stop 21 RCP within 8 hours per S2.OP-AB.RCP-0001.

Answer: B

Explanation / Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because this is an action in S2.OP-AB.RCP-0001 for a leaking or degraded #2 seal.

B. Correct. The conditions in the stem describes a #3 seal leak. Standpipe level low is a primary indicator that the #3 seal is leaking only since the #1 seal leakoff flow is steady at 1.5 gpm which is normal conditions for an intact #2 seal. The crew will take action per the console ARP and perform a makeup to the clear the low standpipe level alarm and initiate S2.OP-ST.RC-0008 for RCS leak rate. The CRS could enter S2.OP-AB.RCP-0001 based on the console alarm, but the AB will not provide any direction for a leaking #3 seal.

C. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse the standpipe level indications in the stem as signs of a #2 seal leak. Part 2 is correct.

D. Incorrect. Part 1 is incorrect. Plausible because the operator may confuse the standpipe level indications in the stem as signs of a #2 seal leak. Part 2 is incorrect. Plausible because this is an action in S2.OP-AB.RCP-0001 for a leaking or degraded #2 seal.

Technical References:	S2.OP-AB.RCP-0001 (R28)
Proposed References to be provided:	N/A
Learning Objective:	NOS05ABRCPUMP-18, Objective 17.b. Given any of the following and appropriate control room reference material, evaluate and determine the effect on the Reactor Coolant Pump: Degraded Reactor Coolant Pump seals and seal water
Question Source:	Modified Bank – Diablo Canyon 2019 NRC SRO86
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	SRO
Category/System:	04-Reactor Coolant Pump System
K/A #:	003 A2.01 Ability to (a) predict the impacts of the following on the Reactor Coolant Pump System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Problems with RCP seals, especially seal leakoff rates
Importance:	4.0
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked to predict the impact of a low RCP standpipe level and based on plant conditions to determine the course of action based on the failed or degraded #3 RCP seal.

Parent Question: Diablo Canyon 2019 SRO86

Modified question to include Part 2 to ask knowledge of what specific action in the AB is required based on the type of seal failure.

Examination Outline Cross-Reference	**Level**	SRO
	Tier #	2
003 A2.01 - 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 with RCP seals, especially rates of seal leak-off	**Group #**	1
	K/A #	003 A2.01
	Rating	3.9

Question 86

Unit 1 is at 100% power.

The following events occur:

- PK 05-01, RCP No.11, alarms, due to input 1259, RCP 1-1 No. 2 Seal Leakoff Flow High
- RCP 1-1 Number 1 seal leakoff is approximately 1.0 gpm
- RCP 1-2, 1-3, 1-4 Number 1 seal leakoff is approximately 3.0 gpm per RCP
- Aux watch reports an increase in RCDT fill rate
- Charging flow is approximately 87 gpm

In accordance with AR PK05-01, what action should be taken by the Shift Foreman?

A. Direct the operator to raise seal injection flow.

B. Direct the operator to trip the reactor, stop RCP 1-1 and enter E-0, Reactor Trip or Safety Injection.

C. Go to OP AP-28, RCP Malfunction, Section B, "RCP Number 1 Seal Failure".

D. Go to OP AP-28, RCP Malfunction, Section C, "RCP Number 2 or 3 Seal Failure".

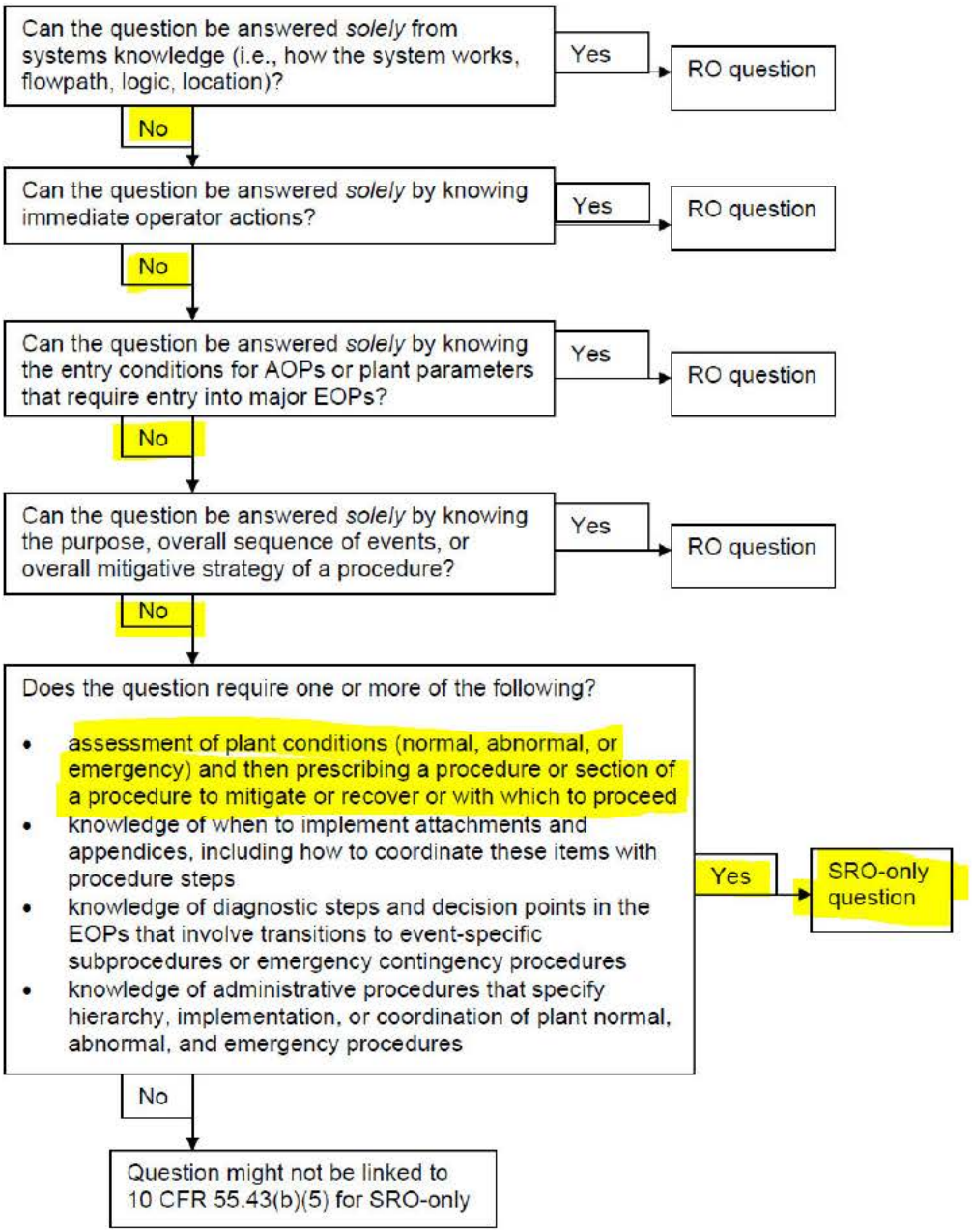
Proposed Answer: D. Go to OP AP-28, RCP Malfunction, Section C, "RCP Number 2 or 3 Seal Failure".

Explanation:

A. Incorrect. Injection flow is lower, but raising it will not address the failure.
B. Incorrect. This is the action in the PK if there are multiple diverse alarms.
C. Incorrect. the 1259 Input directs going to step 2.12 in the AR PK. This step, has actions for both a #1 and #2 or #3 seal failure.This is the action at this step in AR PK for a number 1 seal failure. Indications: Rise in pump radial bearing AND seal outlet temperatures, • Drop in Number 1 Seal DP, • Rise in Number 1 Seal leakoff flow, • Rise in Number 2 Seal leakoff flow, • Rise in Seal injection flow
D. Correct. Lower Number 1 Seal leakoff flow corresponding to the rise in Number 2 Seal leakoff flow, • More frequent filling of Number 3 Seal Standpipe, • Rise in pump vibration, • Recent rise in Containment Sump fill rate

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



87 **10-149** **Points: 1.00**

Given:

- A Loss of Offsite Power has occurred on Salem Unit 2
- The crew is performing actions in 2-EOP-TRIP-4, Natural Circulation Cooldown
- 21 Charging Pump is running
- RCS cooldown to establish required subcooling is in progress
- Low PZR Pressure SI signal is BLOCKED
- RCS temperature is 537 °F

Then:

- RCS pressure is 1183 psig and lowering
- 2CV55, Charging Flow Control Valve, is fully open
- 2CV71, Seal Injection Backpressure Valve, is fully open
- Pressurizer level is 15% and lowering

Which of the following describes the correct action(s) for these conditions per 2-EOP-TRIP-4?

A. Stop the RCS cooldown and wait until PZR level and subcooling are restored.

B. Align the Charging Pump suction to the RWST, open the BIT isolation valves, and transition to 2-EOP-LOCA-1, Loss of Reactor or Secondary Coolant.

C. Operate Pressurizer heaters as necessary to maintain saturated conditions in the pressurizer per 2-EOP-TRIP-4.

D. Actuate Safety Injection and return to 2-EOP-TRIP-1, Reactor Trip or Safety Injection.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because operator may believe that the reason for the PZR level lowering is due to excessive cooldown of the RCS.

B. Incorrect. Plausible because this is similar to CAS step 6.1 in EOP-TRIP-4. Incorrect in that TRIP-4 does not direct entry into EOP-LOCA-1.

C. Incorrect. Plausible because this is similar to CAS step 3 in EOP-TRIP-4. Incorrect in that PZR heaters will not energize with level < 17%.

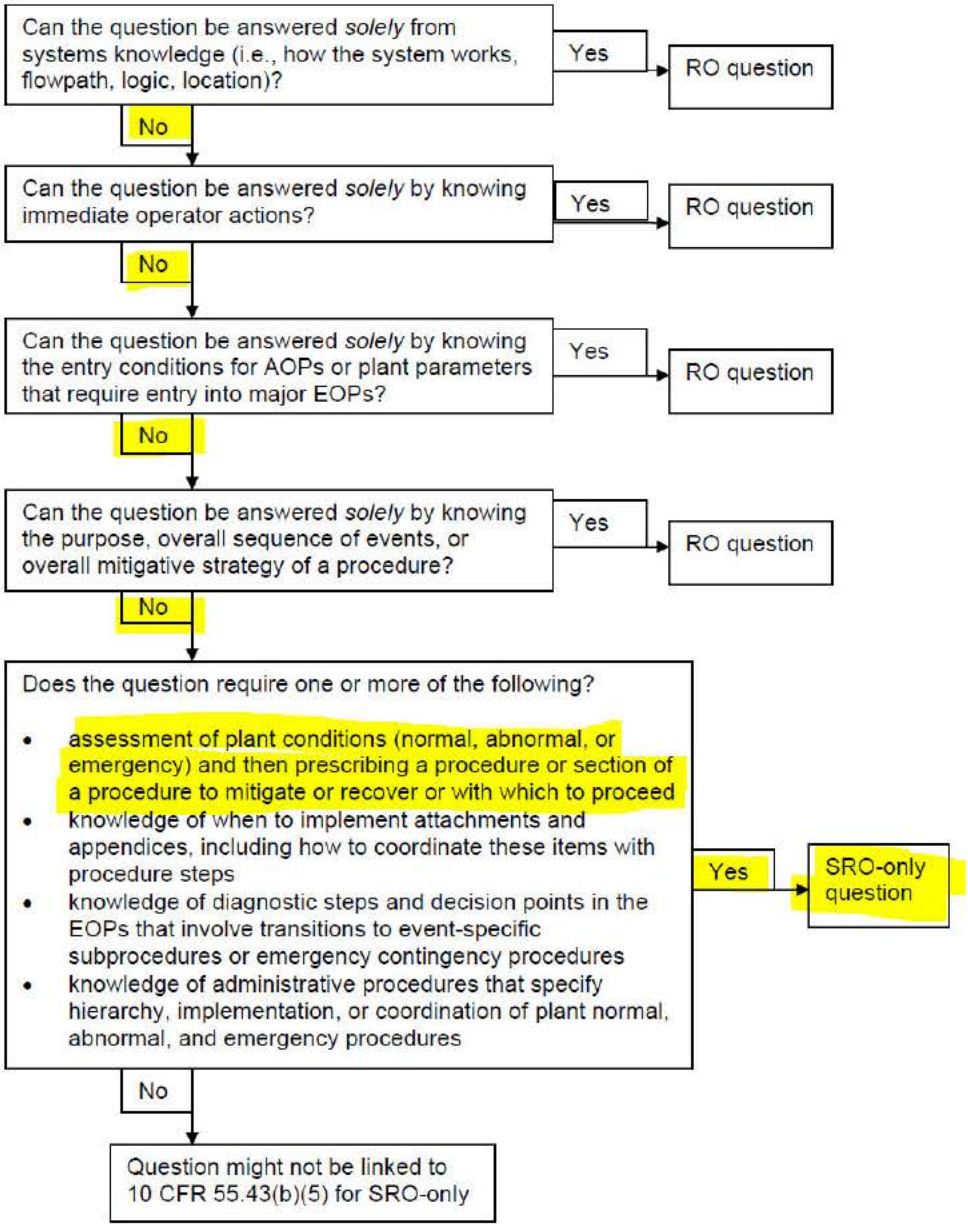
D. Correct. Per the foldout page (CAS) of 2-EOP-TRIP-4, if subcooling is 0 °F or PZR level can NOT be maintained greater than 4%, then Actuate SI and go to 2-EOP-TRIP-1. In this case subcooling is greater than 0 °F, but PZR level is lowering and cannot be maintained based on the 2CV55 and 2CV71 fully open.

Technical References:	2-EOP-TRIP-4 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP004-08, Objective 4. Determine the basis for each step, caution, note or continuous action step relative to a NATURAL CIRCULATION COOLDOWN.
Question Source:	Bank – DC Cook 2008 NRC SRO86
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 45.6
Level:	SRO
Category/System:	02-ECCS
K/A #:	006 A2.12 Ability to (a) predict the impacts of the following on the ECCS system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Conditions requiring actuation of ECCS.
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the candidate is being asked if the conditions in the stem require SI actuation of ECCS equipment.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



88 **06-027** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

Then:

- The Electric System Operator (ESO) reports a Solar Magnetic Disturbance (SMD) of greater than K-5 intensity
- Salem control room has confirmed an Excess MVAR condition

The CRS will direct the crew to…

A. trip the reactor and perform 2-EOP-TRIP-1, Reactor Trip or Safety Injection, and S2.OP-AB.GRID-0001, Abnormal Grid, concurrently.

B. trip the reactor and perform 2-EOP-TRIP-1, Reactor Trip or Safety Injection, Only.

C. initiate load reduction per S2.OP-AB.GRID-0001 Abnormal Grid, and S2.OP-AB.LOAD-0001, Rapid Load Reduction.

D. initiate load reduction per S2.OP-AB.GRID-0001 Abnormal Grid, Only.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because tripping the reactor and going to 2-EOP-TRIP-1 is directed per Attachment 1 if frequency is less than or equal to 57.55 Hz. Incorrect in that performing S2.OP-AB.GRID-0001 is not required to be performed concurrently with 2-EOP-TRIP-1

B. Incorrect. Plausible because the SRO may believe that the grid conditions in the stem require the generator to be immediately isolated from the grid for asset protection and therefore a immediate reactor trip is required.

C. Correct. Per S2.OP-AB.GRID-0001, conditions for SMD are met in step 1.3, the CRS will direct the crew perform Attachment 2, Solar Magnetic Disturbance, and initiate a load reduction to less than or equal to 77.6 % (942 MWe) at 15% per min using S2.OP-AB.LOAD-0001, Rapid Load Reduction.

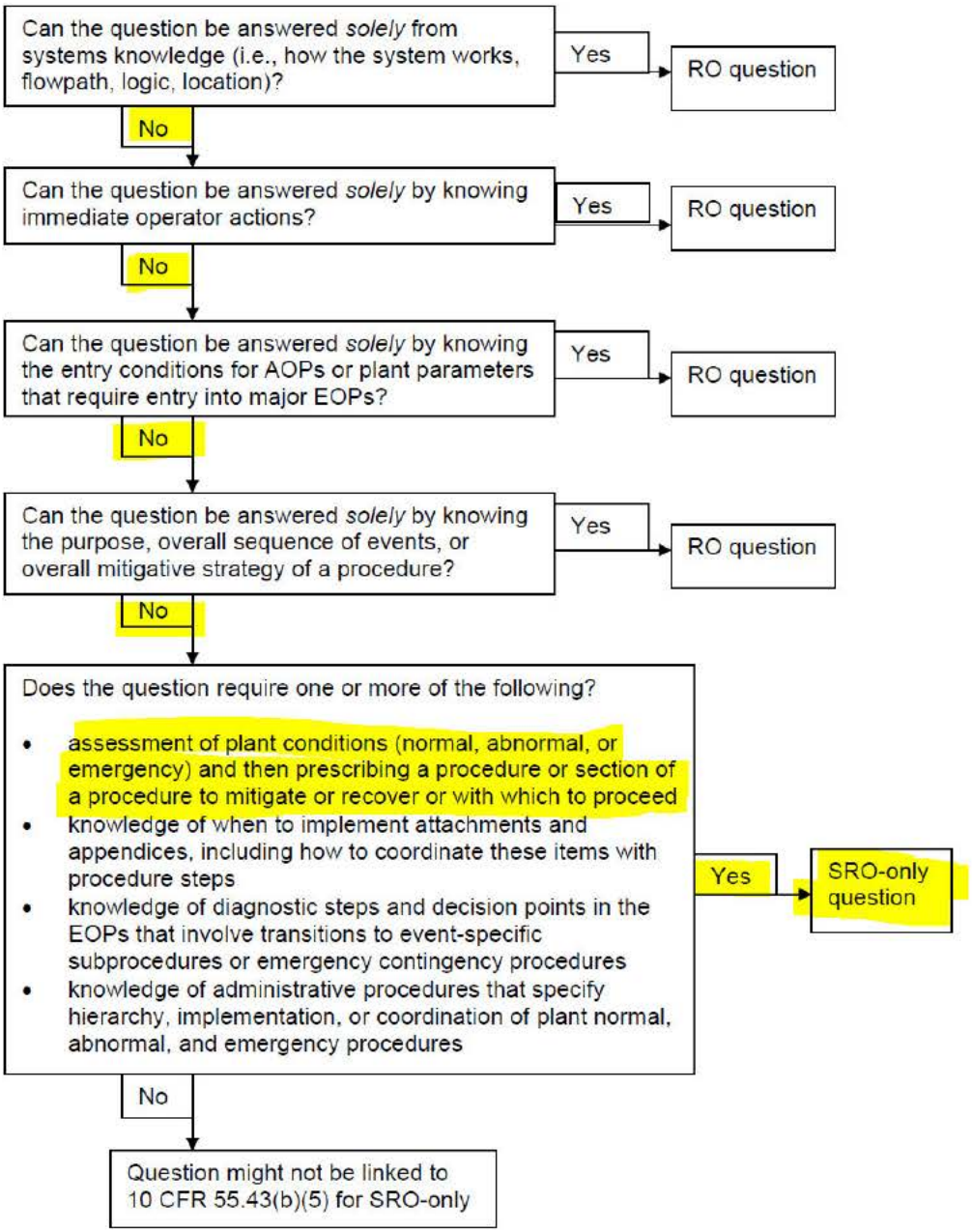
D. Incorrect. Plausible because S2.OP-AB.GRID-0001 does direct initiating a load reduction to 77.6%. Incorrect in that S2.OP-AB.GRID-0001 Attachment 2 also directs initiating the rapid load reduction per S2.OP-AB.LOAD-0001.

Technical References:	S2.OP-AB.GRID-0001 (R23)
Proposed References to be provided:	None
Learning Objective:	NOS05ABGRID-13, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.GRID-0001(Q) and the bases for the actions.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.5 / 45.12 / 45.13
Level:	SRO
Category/System:	06-AC Electrical Distribution System / Conduct of Operations
K/A #:	062 G2.1.6 Ability to manage the control room crew during plant transients (SRO only).
Importance:	4.8
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO how to direct the crew in the control room during a Solar Magnetic Disturbance grid transient per S2.OP-AB.GRID-0001.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



89 **06-028** **Points: 1.00**

Given:

- Salem Unit 1 is at 100% power.
- Preparations are being made for the 1A Emergency Diesel Generator (EDG) surveillance run
- During the pre-job brief, the PO performing the surveillance identified an alarm that is expected to come in numerous times as part of the testing.

Per OP-AA-103-102, Watch Standing Practices, what action will the CRS direct in response to the repetitive alarm?

Direct the PO…

A. that making alarm announcements is NOT required if its associated with the surveillance testing.

B. to bypass the alarm until the test is complete per OP-AA-103-104, Control of Alarm Bypass.

C. that the alarm announcement is still required, but the CRS is NOT required to acknowledge the operator.

D. that ALL alarms associated with surveillance testing must be announced.

Answer: A

Explanation / Justification:

A. Correct. Per OP-AA-103-102 step 4.3.1.3, if repetitive alarms the subsequent announcements are not required with the concurrence of the CRS and RO.

B. Incorrect. Plausible because alarms can be bypassed but the process is intended for malfunctions or inoperable components that are generating the alarm that cannot be cleared until corrective maintenance is performed.

C. Incorrect. Plausible because the SRO may mistakenly confuse the Transient Response process that only requires significant alarms to be announced.

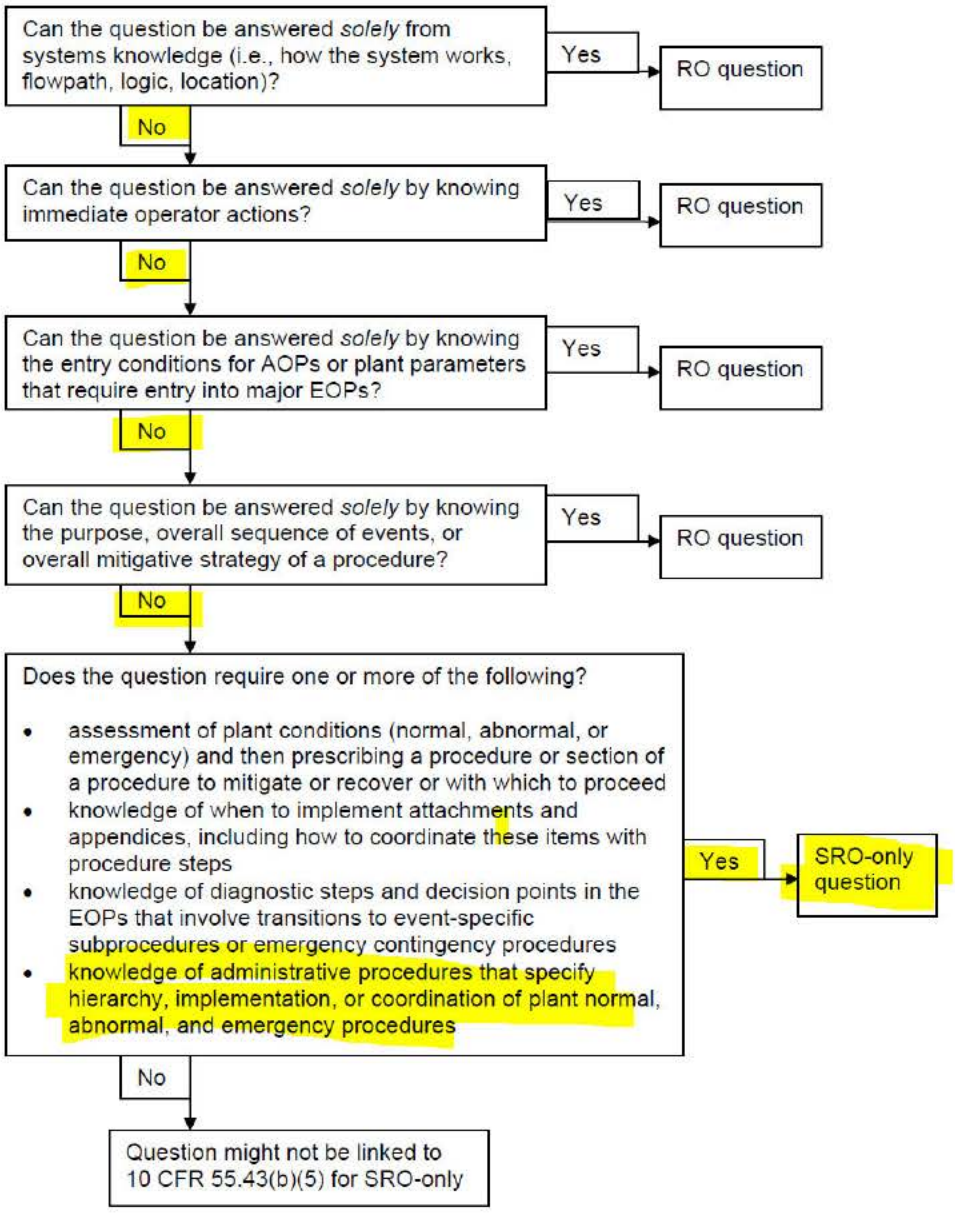
D. Incorrect. Plausible because the SRO may believe that during surveillance testing all alarms need be announced because it could affect operability of the system or component.

Technical References:	OP-AA-103-102 (R13)
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-15 Objective 5.
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	43.1/ 45.5 / 45.12 / 45.13
Level:	SRO
Category/System:	06-Emergency Diesel Generator / Conduct of Operations
K/A #:	064 G2.1.9 Ability to direct licensed personnel activities inside the control room (SRO Only).
Importance:	4.5
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO what action needs to be directed to the licensed PO in the control room with regards to announcing alarms that are expected to be repetitive during a surveillance conducted in the control room.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



90 **07-022** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- A Containment Pressure Relief is in progress

Then:

- 2R12A, Containment Atmosphere Radiation Monitor, fails due to a power supply failure
- The Containment Pressure Relief is stopped

Based on the above condition, which of the following is **ONLY** required per the ODCM to recommence the containment pressure relief (assume all other Radiation Monitors are OPERABLE)?

[REFERENCE PROVIDED]

Per the ODCM requirements, the Containment Pressure Relief…

A. may recommence without any compensatory actions.

B. may recommence provided two independent samples of containment are analyzed and two release rate calculations are independently verified.

C. may recommence provided grab samples are taken at least once per 8 hours and analyzed for principal gaseous emitters.

D. may NOT recommence until 2R12A radiation monitor is restored to OPERABLE status.

Answer: A

Explanation / Justification:

A. Correct. Per Table 3.3-13 (Item 3.a) in 3.3.9 of Salem's ODCM, either 2R12A OR 2R41A & D channels is required. Since 2R41 channels are operable (assumption in stem) then ODCM 3.3.9 is MET and no additional action is required to recommence the containment pressure relief.

B. Incorrect. Plausible because this would be the required action (Action 37) if both 2R12A and 2R41A & D channels are INOPERABLE.

C. Incorrect. Plausible because this would be the required action (Action 33) if the SRO incorrectly selects Item 4.a as the affected ODCM instrument. Incorrect in that this is for the Plant Vent Radiation Monitor (R41).

D. Incorrect. Plausible because this is a required action (Action 34) for a containment purge. Incorrect in that this is for a containment pressure relief.

Technical References:	Salem ODCM 3.3.9 (R29)
Proposed References to be provided:	**ODCM 3.3.9 and Bases**
Learning Objective:	NOS05TECHSPEC-12, Objectives 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12. **(Licensed Operator & STA only)**
Question Source:	Modified Bank – DC Cook 2018 NRC SRO90
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 43.3 / 45.13 / 9
Level:	SRO
Category/System:	07-Prcess Radiation Monitoring System
K/A #:	073 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM System, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PRM component failures
Importance:	3.1
Tier/Group	Plant Systems-Tier 2/Group 1

K/A Match: The K/A is matched because the question is asking the SRO what action is required per the ODCM when the Containment Radiation Monitor fails. The SRO must be able to evaluate the ODCM actions and determine if the pressure relief may continue and any compensatory actions.

Parent Question: DC Cook 2018 NRC SRO90

Modified Question based on Salem's ODCM actions being different for the failure, therefore the correct answer is different.

90 **ID: 2008NRC-0605-SRO** **Points: 1.00**

SRO Only

Given the following conditions on Unit 1:

- The unit is operating at 100% power.
- A Containment Pressure Relief is in progress.
- VRS-1201 Upper Containment Normal Range Monitor failed due to a power supply failure.
- The Containment Pressure Relief is stopped.

Which ONE of the following describes the restrictions placed on Containment Purge/Pressure Relief Operations?

A. Containment Purge/Pressure Relief operations may NOT be performed until VRS-1201 is restored to operable status.

B. Containment Purge/Pressure Relief operations may continue under administrative controls provided that VRS-1201 is restored to operable status prior to entering Mode 4 following the next refueling outage.

C. Containment Purge/Pressure Relief operations may continue under administrative controls for up to 7 days, provided that area surveys of upper containment are performed at least once every 24 hours.

D. Containment Purge/Pressure Relief operations may continue for up to 48 hours before VRS-1201 is required to be restored to operable status.

Answer: B

Answer Explanation:

A. INCORRECT - Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable.

B. CORRECT - Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable. The failed channel must be fixed during the next refueling outage.

C. INCORRECT - Sampling every 24 hours is required per TRM 8.3.8 if BOTH VRS-1101 & 1201 are inoperable. Other Channels within TRM 8.3.8 have a 7 day limit.

D. INCORRECT - Purge operations are not limited. These requirements are from the Technical Specification 3.3.6 Condition C.

Additional References: Salem ODCM

PART I RADIOLOGICAL EFFLUENT CONTROLS

Salem REC ODCM Rev. 29
03/21

3/4.3 INSTRUMENTATION

BASES

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of the "old" 10 CFR Part 20 (ODCM Appendix F). The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

CROSS REFERENCE - TABLES 3.3-13 and 4.3-13

Unit 1 T/S Table Item No.	Instrument Description	Acceptable RMS Channels
1a	Waste Gas Holdup System Noble Gas Activity	1R41A and D(1)(2)
2a	Containment Purge and Pressure - Vacuum Relief Noble Gas Activity	1R12A or 1R41A and D(1)(2)
3a	Plant Vent Header System Noble Gas Activity	1R41A and D(1)(2)
3b	Plant Vent Header System Iodine Sampler (3)	1RME 4, 5 (1R41) or 1RME50, 51 (1R45)
3c	Plant Vent Header System Particulate Sampler (3)	1RME 4, 5 (1R41) or 1 RME50, 51 (1R45)

(1) The channels listed are required to be operable to meet a single operable channel for the ODCM's Minimum Channels Operable" requirement.

(2) 1R41D is the setpoint channel. 1R41A is the measurement channel.

(3) Laboratory analysis of the sampler filters ensures that the limits of ODCM CONTROL 3.11.2.1 are not exceeded. Alarm/trip setpoints do not apply to these passive components.

Additional References: Salem ODCM

3/4.3 INSTRUMENTATION

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE 3.3-13: Radioactive Gaseous Effluent Monitoring Instrumentation

Table Notation

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gaseous principal gamma emitters at the lower limits of detection required in ODCM CONTROL TABLE 4.11-2.A, B, or C within 24 hours. Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that within 4 hours samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, Containment Pressure Reliefs may be performed provided that prior to initiating the release:

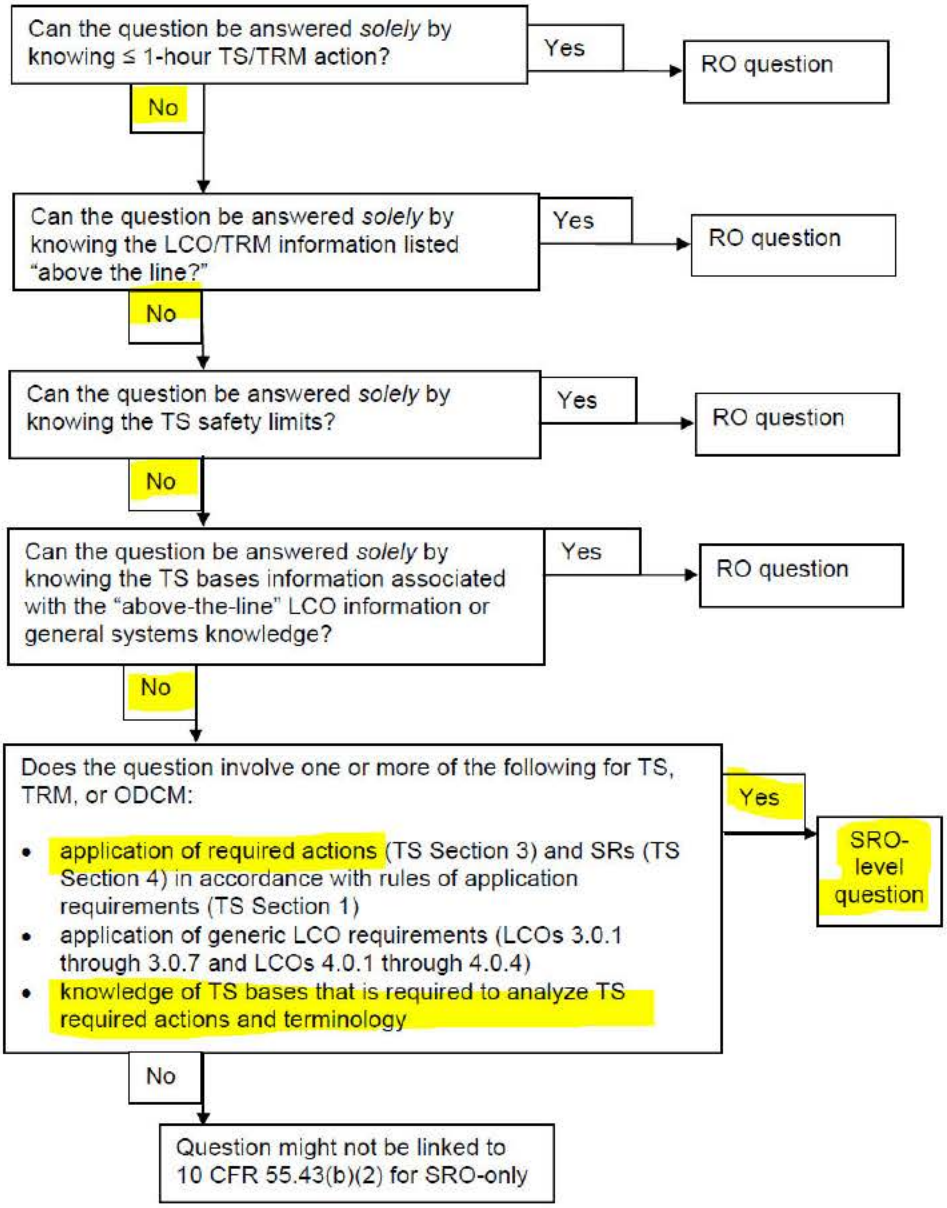
a. At least two independent samples of containment are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations.

Otherwise, suspend release of radioactive effluents via this pathway.

SRO Match Justification:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



91 **02-030** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21 Charging Pump is in service
- Surveillance test S2.OP-ST.RC-0007, Seal Injection Flow, was just completed.

The following test data was recorded:

- Charging Header pressure = 2450 psig
- 2CV55 fully open
- 21 RCP seal injection flow = 10 gpm
- 22 RCP seal injection flow = 10.5 gpm
- 23 RCP seal injection flow = 9.8 gpm
- 24 RCP seal injection flow = 10.2 gpm

Based on the surveillance test data, what action is required?

[REFERENCE PROVIDED]

A. Enter TS 3.5.2, ECCS Subsystems Tavg > 350 °F, for an Inoperable Charging Pump and restore to Operable status within 72 hours.

B. No Tech Spec entry is required the surveillance test is satisfactory.

C. Enter TS 3.5.4, Seal Injection Flow, and adjust manual seal injection throttle valves to within the limit within 4 hours.

D. No Tech Spec entry is required so long as the seal injection flows are adjusted to within the limits within 4 hours.

Answer: C

Explanation / Justification:

A. Incorrect. Plausible because the SRO may incorrectly determine that the unsat seal injection flows is a result of a degraded Charging Pump and therefore declare 21 Charging Pump Inoperable and enter TS 3.5.2 Action a.

B. Incorrect. Plausible if the SRO misinterprets the surveillance data and thinks the test data results meet the Tech Spec surveillance requirements for total seal flows.

C. Correct. The surveillance data for total RCP seal injection flows is Unsat due to exceeding the Tech Spec 3.5.4 limit of ≤ 40 gpm. Tech Spec 3.5.4 is not met and the required action is to make adjustments to restore seal injection flows to within limits within 4 hours.

D. Incorrect. Plausible if the SRO misinterprets the note in Tech Spec 3.5.4 about TS 4.0.4 exemption is allowed for up to 4 hours.

Technical References:	TS 3.5.4 and bases
Proposed References to be provided:	**TS 3.5.2 and 3.5.4**
Learning Objective:	NOS05TECHSPEC-12, Objective 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.2 / 43.5 / 45.12
Level:	SRO
Category/System:	02-Reactor Coolant System
K/A #:	002 G2.2.37 Ability to determine operability or availability of safety related equipment (SRO Only)
Importance:	4.6
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked knowledge on applying TS 3.5.4 with a unsat RCP seal injection surveillance test.

Additional References: Salem TS 3.5.4

EMERGENCY CORE COOLING SYSTEMS

SEAL INJECTION FLOW

LIMITING CONDITION FOR OPERATION

3.5.4 Reactor coolant pump seal injection flow shall be ≤40 gpm with centrifugal charging pump discharge header pressure ≥ 2430 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With seal injection flow not within the limit, adjust manual seal injection throttle valves to give a flow within the limit with the charging pump discharge pressure ≥ 2430 psig and the charging flow control valve full open within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

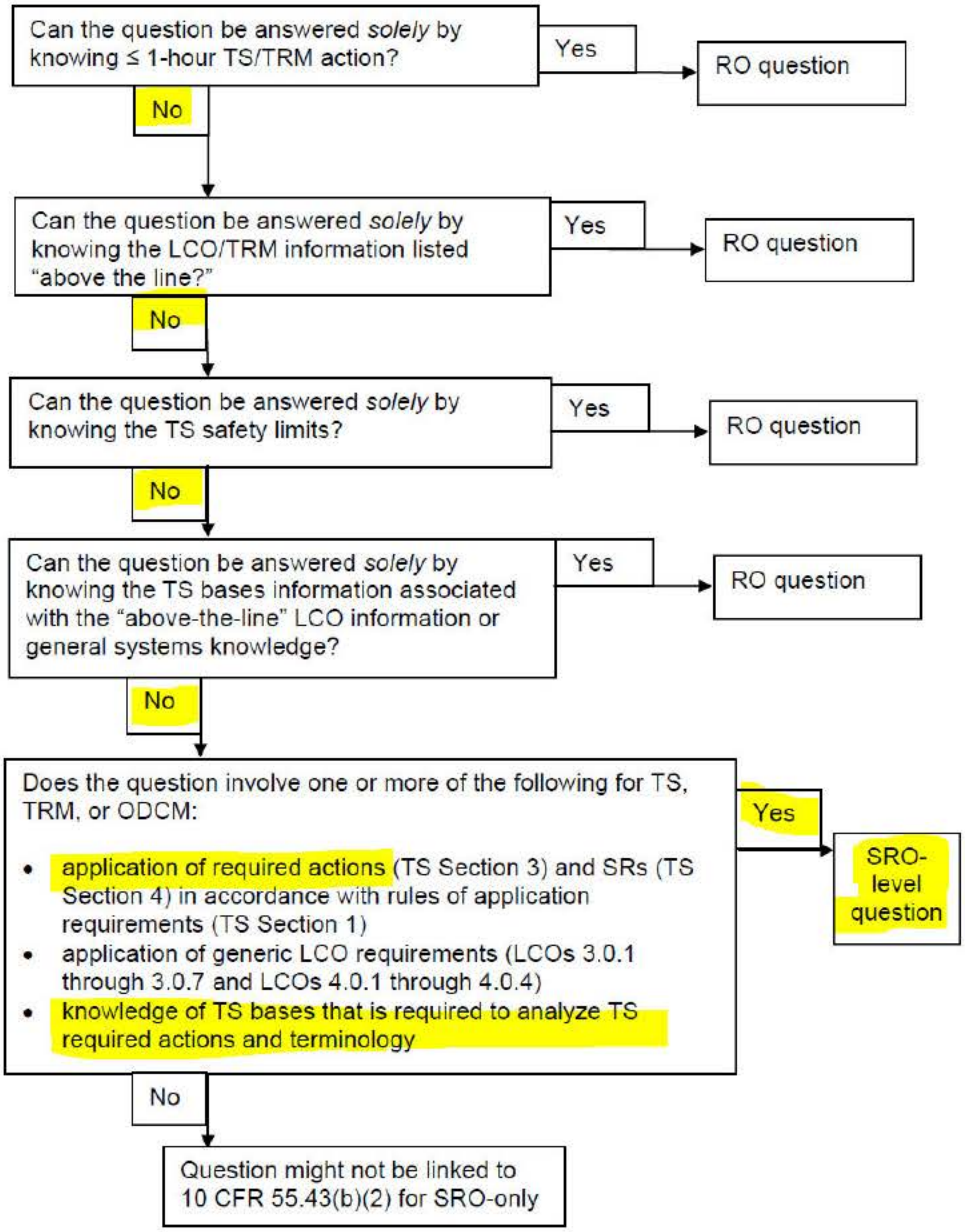
SURVEILLANCE REQUIREMENTS

4.5.4 In accordance with the Surveillance Frequency Control Program, verify manual seal injection throttle valves are adjusted to give a flow within the limit with centrifugal charging pump discharge header pressure ≥ 2430 psig, and the charging flow control valve full open.

The provisions of Specification 4.0.4 are not applicable for entry into Mode 3. This exemption is allowed for up to 4 hours after the Reactor Coolant System pressure stabilizes at 2235 ± 20 psig.

SRO Match Justification:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



92 **07-023** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- PZR Level Channel 1 failed earlier in the shift.
- All actions in S2.OP-AB.CVC-0001, Loss of Charging, and Tech Spec 3.3.1.1 are complete, and the channel has been removed from service per S2.OP-SO.RPS-0003, Placing Pressurizer Channel in Tripped Condition.
- PZR Level Channel 3 has been selected for control and Channel 2 for alarm.
- PZR Level is on program.

Then:

- PZR Level Channel 3 differential pressure transmitter sensor fails due to the internal bellows rupturing

Which of the following completes both statements?

1. PZR Level Channel 3 indication will be __(1)__.
2. What procedure will the CRS direct to respond to this event?

A. (1) lower
(2) 2-EOP-TRIP-1, Reactor Trip or Safety Injection

B. (1) lower
(2) S2.OP-AB.CVC-0001, Loss of Charging

C. (1) higher
(2) S2.OP-AB.CVC-0001, Loss of Charging

D. (1) higher
(2) 2-EOP-TRIP-1, Reactor Trip or Safety Injection

Answer: D

Explanation / Justification:

A. Incorrect. First part is incorrect. Plausible if the operator incorrectly recalls how the bellows failure will affect the differential pressure sensed by the transmitter. Second part is correct.

B. Incorrect. First part is incorrect. Plausible if the operator incorrectly recalls how the bellows failure will affect the differential pressure sensed by the transmitter. Second part is incorrect. Plausible if the operator believes that the reactor will not trip on PZR level high (>92%) because with channel 1 properly removed from service this channel is bypassed to a 2/2 logic for PZR level high.

C. Incorrect. First part is correct. Second part is incorrect. Plausible if the operator believes that the reactor will not trip on PZR level high (>92%) because with channel 1 properly removed from service this channel is bypassed to a 2/2 logic for PZR level high.

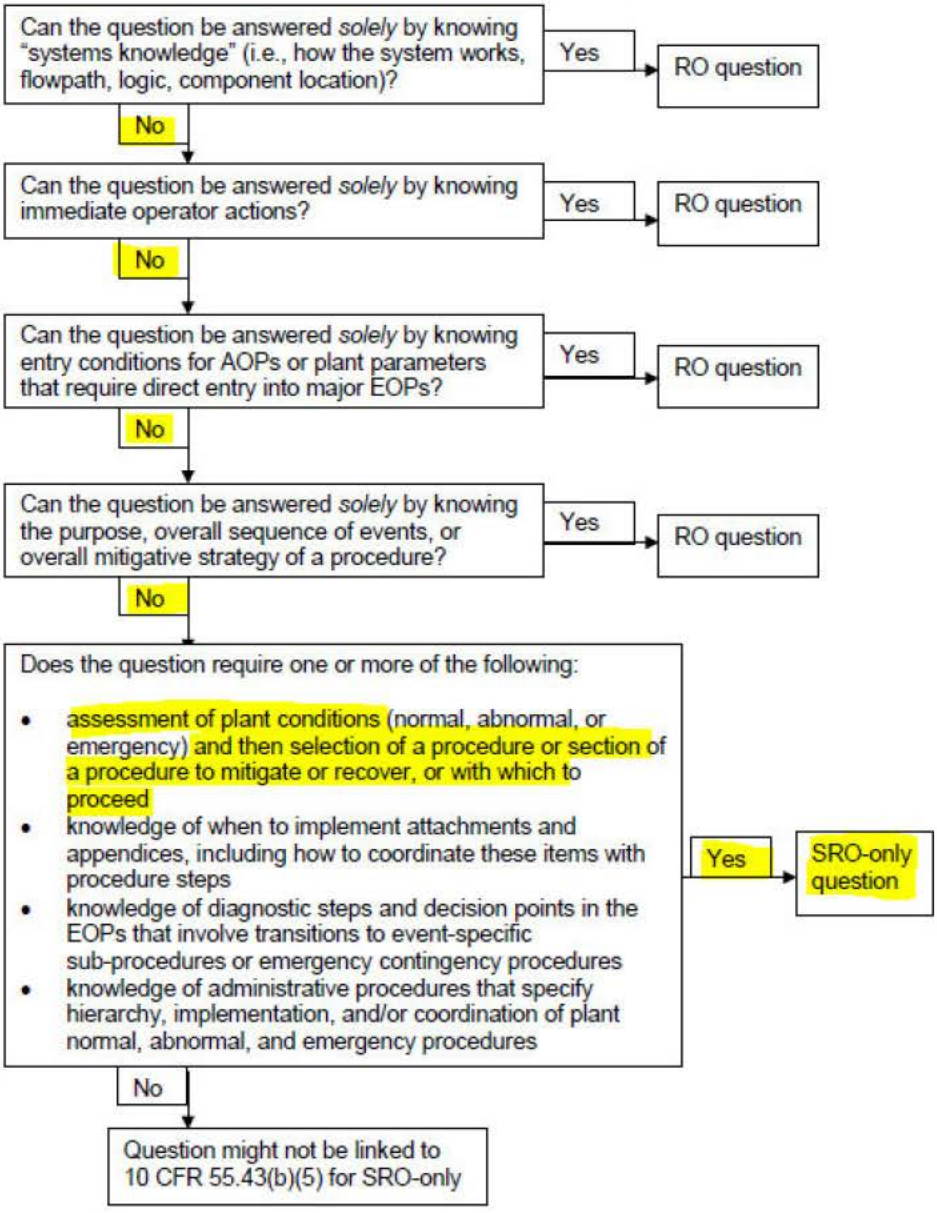
D. Correct. A bellows rupture will cause the PZR level channel 3 to fail high because the transmitter will sense a D/P of zero indicating the PZR is full. Therefore, with all actions for channel 1 failing low being complete, the bistables for channel 1 are tripped. With 2/3 PZR level indicators being > 92%, the reactor will automatically trip. Proper procedure for this reactor trip is EOP-TRIP-1.

Technical References:	S2.OP-AB.CVC-0001 (R14), INPO Sensors and Detectors
Proposed References to be provided:	None
Learning Objective:	NOS05SENSOR-03.
Question Source:	Bank - Salem
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 /45.6
Level:	SRO
Category/System:	07-Nonnuclear Instrumentation System
K/A #:	016 A2.01 Ability to (a) predict the impacts of the following on the Nonnuclear Instrumentation System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector/transmitter failure
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked to predict the impact on PZR level when the bellows in the transmitter fails and based on the transmitter failure select the appropriate procedure to address the plant impact.

SRO Match Justification:

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



93 **08-020** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- 21A CW Pump is C/T for waterbox cleaning

Then:

- The 4KV Infeed breaker to the 24 CW bus fails opens

What action will the CRS direct the crew?

A. Power operation at 100% may continue since the 24 CW Bus will auto transfer to the 23 CW Bus.

B. Initiate a load reduction to less than 49% power and trip the main turbine per S2.OP-AB.TRB-0001, Turbine Trip Below P-9.

C. Initiate a load reduction to less than or equal to 83% power per S2.OP-AB.CW-0001, Circulating Water Malfunction.

D. Trip the reactor and enter 2-EOP-TRIP-1, Reactor Trip or Safety Injection, per S2.OP-AB.CW-0001, Circulating Water Malfunction.

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the SRO may believe that the CW Bus Auto Transfer scheme will transfer the 24 CW bus to 23 CW bus without losing power to the 'B' Circulators and therefore, no impact to the current power level.

B. Incorrect. Plausible because the SRO may believe that removing the main turbine from service is allowed using S2.OP-AB.TRB-0001.

C. Incorrect. Plausible because a loss of both 21A and 22A circulators requires a load reduction to less than or equal to 83% power.

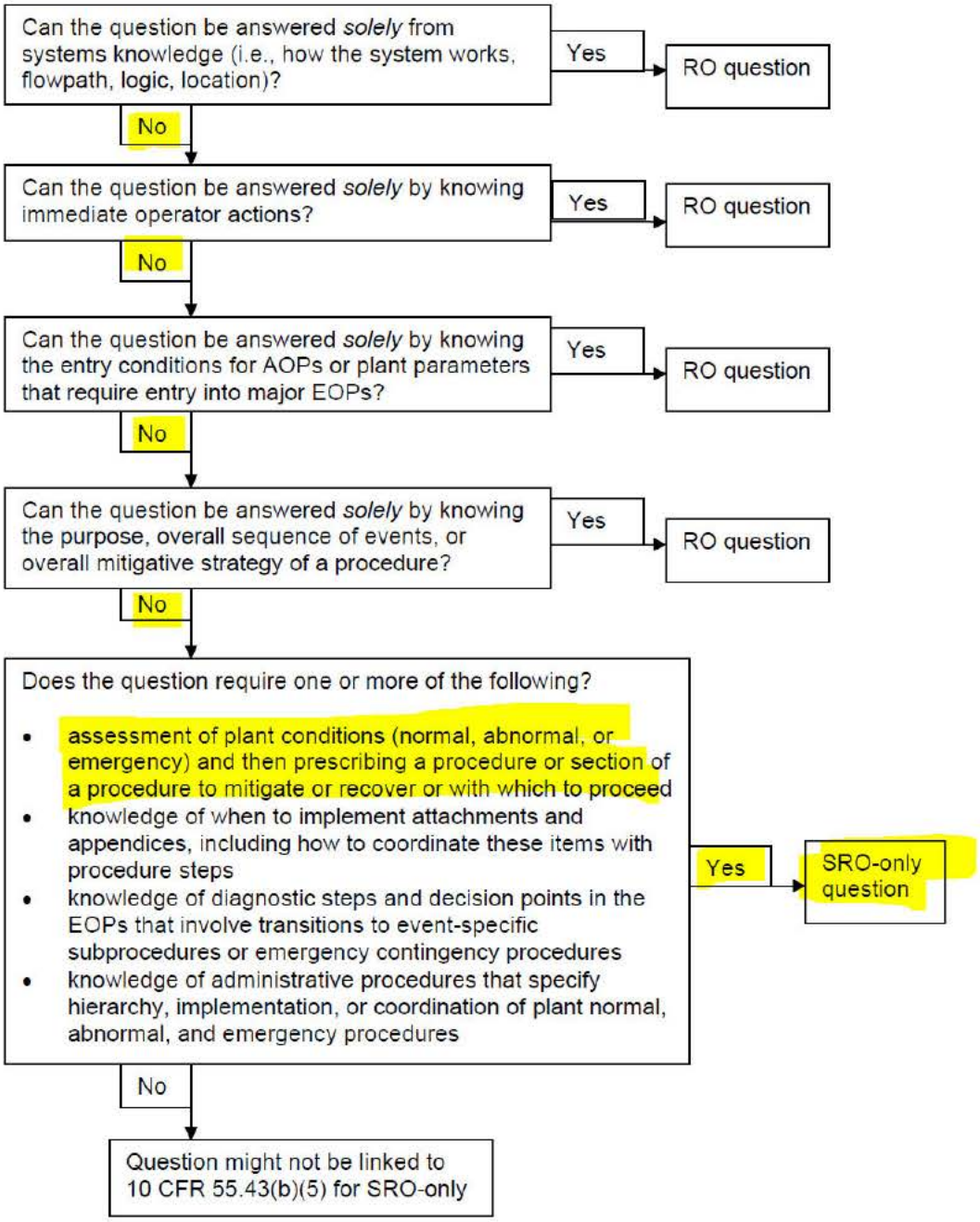
D. Correct. When the infeed breaker (24CW9AD) fails open, the 24 CW bus will de-energize, when the bus voltage is < 70% the bus will transfer automatically to the 23 CW bus. However, all the 'B' Circulator (21B, 22B, and 23B) breakers will trip open on undervoltage and must be manually started following the bus transfer. The traveling screen power will be supplied from the 23 CW bus providing power to the 'A' Circulators. Per S2.OP-AB.CW-0001 Attachment 1, with 4 or more circulators out of service with power greater than or equal to P-10 (10%), then trip the reactor and enter 2-EOP-TRIP-1.

Technical References:	S2.OP-AB.CW-0001 (R41)
Proposed References to be provided:	None
Learning Objective:	NOS05ABCW01-13, Objective 4. Describe, in general terms, the actions taken in S2.OP-AB.CW-0001 and the bases for the actions in accordance with the Technical Bases Document.
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.3 / 45.13
Level:	SRO
Category/System:	08-Circulating Water System
K/A #:	075 A2.02 Ability to (a) predict the impacts of the following on the CW System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of circulating water pumps
Importance:	3.4
Tier/Group	Plant Systems-Tier 2/Group 2

K/A Match: The K/A is matched because the SRO is being asked to assess plant conditions and then select the specific procedural to mitigate the loss of 4 or more Circulators.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



3.11 Is the Containment Equipment Hatch open?

___ NO ___ YES ——> **GO TO** Step 3.13

Time

___ 3.12 **ISOLATE** Containment as follows:

___ A. **INITIATE** S2.OP-AB.CONT-0001(Q), Containment Closure.

___ B. **GO TO** Step 3.14.

___ 3.13 **ISOLATE** Containment as follows:

NOTE

Either the Containment Purge system or the Auxiliary Building Ventilation System, with suction from the containment atmosphere, with the associated radiation monitoring will be available whenever movement of irradiated fuel is in progress in the Containment and the Containment Equipment Hatch is open. This is to ensure airflow is into the Containment and that radiation monitoring remains available.

___ A. **ENSURE** one of the following:

___ 1. A Containment Purge is in progress IAW S2.OP-SO.WG-0006(Q), Containment Purge to the Plant Vent.

OR

___ 2. ABV is in service IAW S2.OP-SO.ABV-0001(Q), Auxiliary Building Ventilation System Operation, with at least one personnel airlock open between Containment and the Auxiliary Building.

NOTE

- Either the Containment Purge shall remain inservice OR at least one personnel airlock shall remain open, with ABV in service, to ensure airflow into the Containment and radiation monitoring remains available while the equipment hatch is open.
- Containment Closure shall be accomplished within 1 hour IAW S2.OP-AB.CONT-0001(Q).

___ B. **INITIATE** S2.OP-AB.CONT-0001(Q), Containment Closure.

94 **14-082** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- The crew is at minimum shift complement

Then:

- A shift change is in progress
- An on-duty Unit 2 Reactor Operator reports that his relief is unable to report to work
- There are NO available qualified licensed ROs on-site

In accordance with OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, which of the following completes both statements?

1. The on-duty Unit 2 RO shift position __(1)__ be unmanned upon shift change.
2. The shift complement may be one less than the minimum requirements for a period not to exceed __(2)__.

A. (1) can (2) 2 hours

B. (1) can (2) 1 hour

C. (1) can NOT (2) 1 hour

D. (1) can NOT (2) 2 hours

Answer: D

Explanation / Justification:

A. Incorrect. Plausible because the SRO may misinterpret the requirements in OP-AA-101-111 and believe that since 2 hours is allowed to fill the unexpected absence that the on duty RO can leave at the end of the shift.

B. Incorrect. Plausible because the SRO may misinterpret the requirements in OP-AA-101-111 and believe that since 2 hours is allowed to fill the unexpected absence that the on duty RO can leave at the end of the shift.

C. Incorrect. Plausible because 1 hour requirements are used in Tech Specs and the SRO may believe that this 1 hour action is warranted.

D. Correct. Per OP-AA-101-111, Attachment 2 Note 1, the Salem Shift complement may be one less than the minimum requirements for a period not to exceed 2 hours to accommodate the unexpected absence of on duty shift members provided that immediate action is taken to restore the Shift Complement to within the minimum requirements. Also, this provision does not permit any shift position to be unmanned upon shift change due to an oncoming shift member being late or absent. The preferred action is to hold the off-going shit personnel until a relief is available.

Technical References:	OP-AA-101-111 (R11), TS 6.2.2
Proposed References to be provided:	None
Learning Objective:	NOS05CONDOP-14, Objective 5.g. Describe requirements for the following Control Room or Field Activities in accordance with applicable Conduct of Operations Manual Administrative: Procedures: Shift Relied and Turnovers
Question Source:	New
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.12
Level:	SRO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential).
Importance:	3.9
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the candidate requires knowledge of administrative procedures related to minimum shit staffing requirements. The question is asking the SRO the required time to fill a vacant licensed position per TS and whether the on duty shift personnel can leave if his relief is absent.

Additional Reference:

Attachment 2, Salem Shift Complement

[CD 419Y, 70145753, T/S 6.2.2] (Notes 1, 6, and 7)

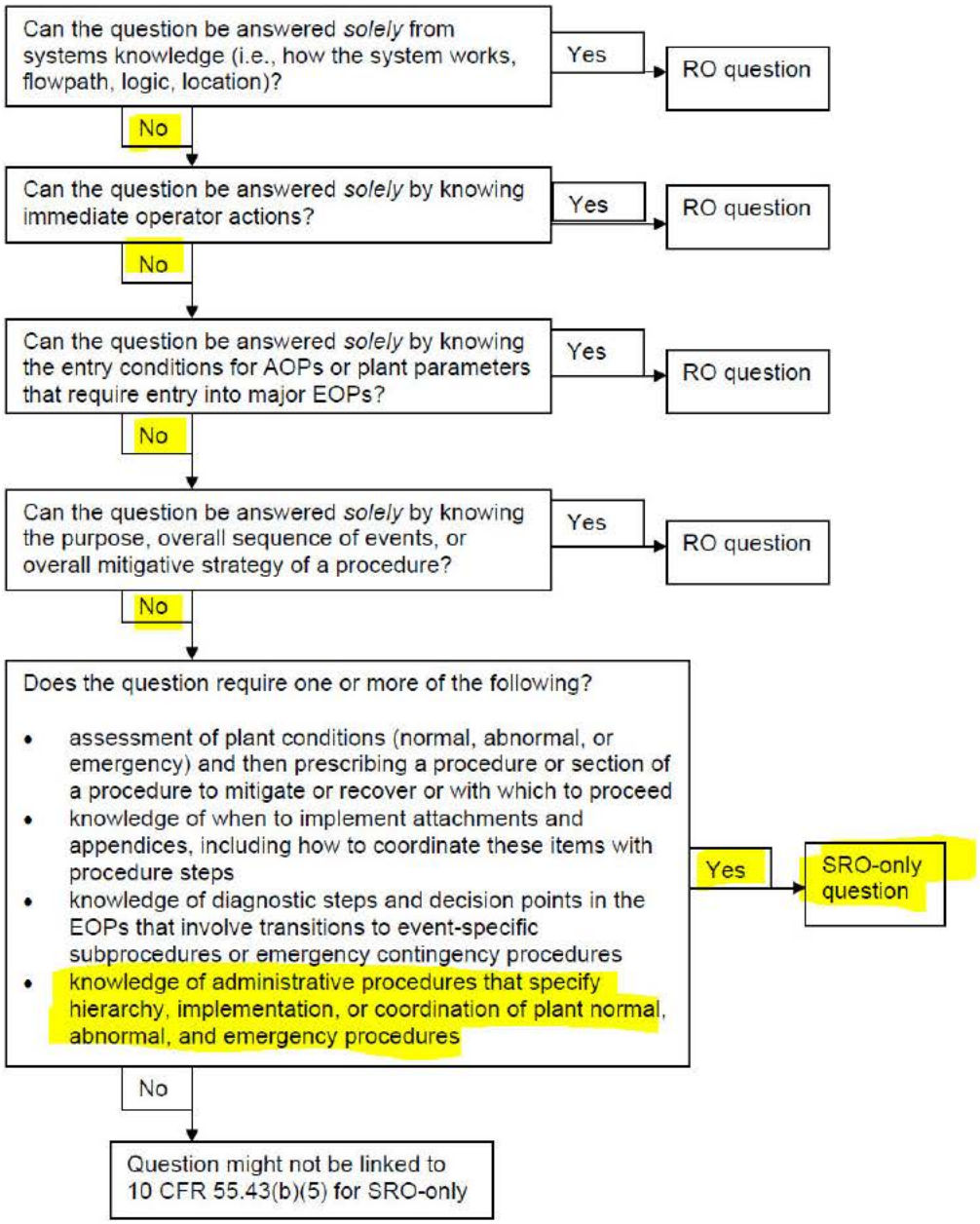
	Salem Unit 1 Mode	**1-4**	**1-4**	**5-6, D**	**5-6, D**
	Salem Unit 2 Mode	**1-4**	**5-6, D**	**1-4**	**5-6, D**
	SM	1	1	1	1
	CRS	2	2	2	1
	STA	1	1	1	1
	OSCC (Note 4)	1	1	1	1
	RO/PO	4	4	4	4
	NEO	6	6	6	6
	Communicator (Note 2)	2	2	2	2
CD-252X	Shift Electrician	1 (each)	1 (each)	1 (each)	1 (each)
CD-252X	Shift I&C Technician	1 (each)	1 (each)	1 (each)	1 (each)
CD-252X	RP Technician (Note 3)	2	2	2	2
	Fire Brigade Liaison (FBL) (Note 8)	1	1	1	1
	Fire Brigade	IAW FP-AA-012			
	Debris Removal Operator (DRO) (Note 5)	1	1	1	1
	Towing Operator (TO) (Note 5)	1	1	1	1

This attachment combines the minimum shift staffing requirements of the Technical Specifications, Emergency Planning, Appendix R Simultaneous Alternate Shutdown, and Beyond Design Basis External Event (BDBEE) FLEX Strategies. Clarifying statements are identified separately below:

1. Except for the SM, the Salem Shift Complement may be one less than the minimum requirements for a period not to exceed 2 hours to accommodate the unexpected absence of on duty shift members provided that immediate action is taken to restore the Shift Complement to within the minimum requirements of the tables. This provision does not permit any shift position to be unmanned upon shift change due to an oncoming shift member being late or absent. The preferred action is to hold over the off-going shift personnel until a relief is available. **[CD 252X]**

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



95 **14-083** **Points: 1.00**

Given:

- Salem Unit 2 is in MODE 6
- Conditions are established per S2.OP-IO.ZZ-0010, Spent Fuel Movements, to support fuel handling operations in the Fuel Handling Building (FHB)
- Movement of irradiated fuel in the FHB is in progress

Then:

- FHB console alarm, BLDG AIR D/P LO, is Alarming
- No operator action has been taken

Based on the above condition and per S2.OP-AR.ZZ-0011, 2CC1 Console Alarm, which of the following completes both statements?

1) What is the NEXT required action?
2) If the BLDG AIR D/P LO alarm is INVALID and will NOT clear, then the local indicator __(2)__ be used to continuously monitor FHB D/P.

A. (1) Validate the alarm using the local indicator.
(2) may NOT

B. (1) Suspend movement of irradiated fuel assemblies.
(2) may

C. (1) Suspend movement of irradiated fuel assemblies.
(2) may NOT

D. (1) Validate the alarm using the local indicator.
(2) may

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because validating the console alarm is an action performed. Incorrect in that the first action is to suspend fuel moves in the FHB and then validate the alarm.

B. Correct. Per S2.OP-AR.ZZ-0011 for BLDG AIR D/P LO, if movement of irradiated fuel assemblies are in progress, then suspend movement of irradiated fuel assemblies in the FHB. The next action is to validate the alarm by checking the local indicator. If the alarm is considered invalid, then compensatory actions are required. Per S2.OP-IO.ZZ-0010, if the BLDG AIR D/P LO is not clear, then the local indicator may be used to continuously monitor FHB D/P.

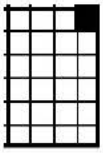
C. Incorrect. Plausible because the SRO may believe that without the console alarm operable, then fuel moves inside the FHB cannot continue until the alarm function is restored. Incorrect in that procedures allow the use of the local indicator so long as its continuously monitored.

D. Incorrect. Plausible because validating the console alarm is an action performed. Incorrect in that the first action is to suspend fuel moves in the FHB and then validate the alarm.

Technical References:	S2.OP-IO.ZZ-0010 (R38), S2.OP-AR.ZZ-0011 (R64)
Proposed References to be provided:	None
Learning Objective:	NOS05IOP010-05, Objective 7. Evaluate the actions required by TSs or TRM during performance of S1/S2.OP-IO.ZZ-0010(Q), Spent Fuel Pool Manipulations if any of the following limits are exceeded: FHB DP greater than -0.125 (less negative).
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	43.7
Level:	SRO
Category/System:	14-Conduct of Operations
K/A #:	G2.1.35 Knowledge of the fuel handling responsibilities of SROs (SRO Only).
Importance:	3.9
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO is being asked knowledge of fuel handling responsibilities associated with the movement of irradiated fuel in the FHB and what action is required when the FHB building D/P alarm is in and not valid.

Additional Reference: S2.OP-AR.ZZ-0011 (2CC1)

SYSTEM	FUEL HANDLING BUILDING VENTILATION Exhaust Filter Trains	BLDG AIR D/P LO
		
BEZEL	Controls	

DEVICES: 2PD-6531

SETPOINT: -0.160" H_2O

1.0 **CAUSE(S)**:

1.1 Loss of slight negative pressure in the Fuel Handling Building

2.0 **AUTOMATIC ACTIONS**:

2.1 None

3.0 **OPERATOR ACTIONS**:

3.1 IF movement of irradiated fuel assemblies is in progress
THEN **PERFORM** the following:

A. **SUSPEND** movement of irradiated fuel assemblies in the FHB.

B. **SEND** an Operator to Panel 436-2 (130' FHB):

1. **CHECK** alarm setpoint at 2PD6531, FHB VENT LO DP SW.

2. **VALIDATE** the BLDG AIR D/P LO alarm using local indicator 2PD6531.

C. IF alarm is validated, **DEPRESS** INLET VANE OPEN bezel for 21 AND 22 FHB EXHAUST FANS.

D. IF alarm clears OR FHB can be maintained at a slight negative pressure
THEN movement of irradiated assemblies may continue.

E. IF alarm remains illuminated with FHB at negative pressure
THEN **IMPLEMENT** compensatory actions as required.

Additional Reference:

S2.OP-IO.ZZ-0010(Q)

ATTACHMENT 3
(Page 1 of 3)

Sheet ___ of ___

REVIEWS FOR CONTINUING FUEL HANDLING

1.0 SHIFT REVIEW:

1.1 **REVIEW** entire procedure
AND **ENSURE** ALL conditions still valid.

1.2 **ENSURE** the following:
A. BOTH FHB Exhaust Fans AND one Supply Fan must be AVAILABLE AND operating.
B. Fuel Handling Building is maintained at slight negative pressure with respect to atmosphere pressure.
C. BLDG AIR D/P LO alarm is clear OR local indicator 2PD6531 being used to continuously monitor FHB ΔP.

1.3 **ENSURE** ALL Open Items listed in Section 2.0 of attachment resolved prior to resuming Spent Fuel Pool Manipulations.

SRO Match Justification:

f. *Procedures and Limitations Involved in Initial Core Loading, Alterations in Core Configuration, Control Rod Programming, and Determination of Various Internal and External Effects on Core Reactivity [10 CFR 55.43(b)(6)]*

Some examples of SRO-only examination items for this topic include the following:

- evaluation of core conditions and emergency classifications based on core conditions
- administrative requirements associated with low-power physics testing processes
- administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths or activities
- administrative controls associated with the installation of neutron sources
- knowledge of TS bases for reactivity controls

g. *Fuel Handling Facilities and Procedures [10 CFR 55.43(b)(7)]*

Some examples of SRO-only examination items for this topic include the following:

- refuel floor SRO responsibilities
- assessment of fuel handling equipment SR acceptance criteria
- prerequisites for vessel disassembly and reassembly
- decay heat assessment
- assessment of SRs for the refueling mode
- reporting requirements
- emergency classifications

96 **14-084** **Points: 1.00**

Per WC-AA-101, On-Line Work Management Process, which of the following choices are responsibilities of the Shift Manager in support of online maintenance?

1. Ensures fire in a(4) Risk Management Actions are implemented.
2. Ensures risk is reassessed when emergent conditions results in a plant configuration not previously assessed.
3. Ensures Protected Equipment barriers are applied per OP-AA-108-116, Protected Equipment Program".
4. Manages special or complex projects during on-line implementation.

A. 1, 2, 3, and 4

B. 1, 2, and 3 Only

C. 2 and 3 Only

D. 1 and 4 Only

Answer: B

Explanation / Justification:

A. Incorrect. Plausible because the SRO may believe that the SM has this responsibility based on the complex project. Incorrect in that per WC-AA-101, the responsibility belongs to the Online Maintenance Project Coordinator.

B. Correct. Per WC-AA-101, step 3.6, the Shift Manager responsibilities include the following: 3.6.1 Ensures shift operations reassesses risk if an emergent condition results in a plant configuration that has not been previously assessed, 3.6.2 Ensures shift operations apply Protected Equipment barriers IAW OP-AA-108-116, 3.6.3 Ensures fire in a(4) Risk Management Actions are implemented as applicable IAW the appropriate site specific procedure, 3.6.4 Ensures the site F:EX capability is maintained.

C. Incorrect. Plausible because these are responsibilities of the SM. Incorrect in that choice 1 is also a responsibility of the SM.

D. Incorrect. Plausible because choice 1 is a responsibility of the SM. Incorrect in that per WC-AA-101, the responsibility belongs to the Online Maintenance Project Coordinator.

Technical References:	WC-AA-101 (R30)
Proposed References to be provided:	None
Learning Objective:	NOS05WORK00-07, Objective4. Describe the conditions under which the SM/CRS permission must be obtained prior to starting work, in accordance with WC-AA-10, Work Management Process Description, and WC-AA-101, On-Line Work Management Process.
Question Source:	Modified - Byron 2017 NRC SRO96
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	14-Equipment Control
K/A #:	G2.2.19 Knowledge of maintenance work order requirements.
Importance:	3.4
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the question is asking the SRO knowledge on administrative controls for performing maintenance on systems and plant components per WC-AA-101.

Additional Reference: WC-AA-101 page 6

3.4 **Work Week Manager (WWM)**

3.4.1. Ensures that a risk assessment of planned work has been performed and evaluates the results against the criteria of Attachment 3, Configuration Management Criteria

3.4.2. Oversees the preparation and coordination activities of Project Coordinators required by this procedure.

3.4.3. Ensures completion and approval of Attachment 2, On-Line Maintenance Approval Form, when required.

3.5 **Online Maintenance Project Coordinator**

3.5.1. Designated by Cycle Manager or WWM to manage special or complex projects during on-line operation.

3.5.2. Serves as the single point of contact between the Shift, Work Week Manager, and work groups.

3.6 **Shift Manager**

3.6.1. Ensures shift operations reassesses risk if an emergent condition results in a plant configuration that has not been previously assessed.

3.6.2. Ensures shift operations apply Protected Equipment barriers IAW OP-AA- 108-116, "Protected Equipment Program".

3.6.3. Ensures fire in a(4) Risk Management Actions are implemented as applicable IAW the appropriate site specific procedure.

3.6.4. Ensures the site FLEX capability is maintained.

3.7 **Operations Services Manager**

3.7.1. Ensures any threats to generation are provided to the Electrical System Operations Center (ESOC) as prescribed by OP-AA-108-107-1001, Electric System Emergency Operations and Electric Systems Operator Interface.

3.7.2. Ensures planned load reductions are coordinated with the ESOC.

Parent Question: Byron 2017 NRC SRO96

Modified question by adding a third correct choice and some of the choices from the parent question are different that PSEG site procedure.

96 **ID: 2017 NRC Q96C** **Points: 1.00**

Per WC-AA-101, ONLINE WORK CONTROL PROCESS, which of the following are responsibilities of Operations Shift Management?

1. Audits the work planning and scheduling processes to ensure maintenance is completed.
2. Ensures appropriate actions are taken to mitigate online risk.
3. Responsible for declaring Structure, System, and Component (SSC) functionality/availability.
4. Ensures project critiques are performed for projects requiring a fragnet.

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 4 and 1

Answer: B

SRO Match Justification:

c. *Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]*

Some examples of SRO-only examination items for this topic include the following:

ES-4.2, Page 21 of 29

- screening and evaluation processes under 10 CFR 50.59, "Changes, tests, and experiments"
- administrative processes for temporary modifications
- administrative processes for disabling annunciators
- administrative processes for the installation of temporary instrumentation
- processes for changing the plant or plant procedures

97 **14-085** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power

At Time 0500:

- The crew enters S2.OP-AB.RC-0001, RCS Leak, for a suspected RCS leak

At Time 0505:

- The estimated leak rate is 12 gpm
- The leak is identified from the 2CV6, Letdown Relief Valve

At Time 0516:

- Letdown is isolated and the leakage has stopped

Which of the following completes both statements?

1. At time 0505, what is the status of TS 3.4.7.2, Operational Leakage?
2. At time 0516, the CRS _____ required to make an Emergency Declaration.

[REFERENCE PROVIDED]

A. (1) NOT Met
(2) is NOT

B. (1) NOT Met
(2) is

C. (1) Met
(2) is

D. (1) Met
(2) is NOT

Answer: D

Explanation / Justification:

A. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is correct.

B. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is incorrect. Plausible because the SRO may incorrectly determine that the RCS leakage exceeded the criteria EAL SU5.1.

C. Incorrect. Part 1 is incorrect. Plausible because the SRO may believe that the CVCS system is part of the RCS system. Incorrect in that the CVCS system is considered an auxiliary system and not part of the RCS system. Part 2 is incorrect. Plausible because the SRO may incorrectly determine that the RCS leakage exceeded the criteria EAL SU5.1.

D. Correct. Per the bases of S2.OP-AB.RC-0001, a RCS leak on the CVCS system that can be isolated does not require entry into TS 3.4.7.2 or requires an Emergency Notification (UE EAL SU5.1).

Technical References:	S2.OP-AB.RC-0001 (R18), EP-SA-325-218 (R0)
Proposed References to be provided:	**EP-SA-325-218 (Bases) and EP-SA-325-118 (EAL SU5.1)**
Learning Objective:	NOS05ABRC01-04, Objective 2. Describe, in general terms, the actions taken in S1/S2.OP-AB.RC-0001 and the bases for the actions in accordance with the Technical Bases Document
Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10CFR Part 55 Content:	41.5 / 43.5 / 45.12
Level:	SRO
Category/System:	14-Equipment Control
K/A #:	G2.2.44 Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.
Importance:	4.4
Tier/Group	Generic Knowledge and Abilities-Tier 3

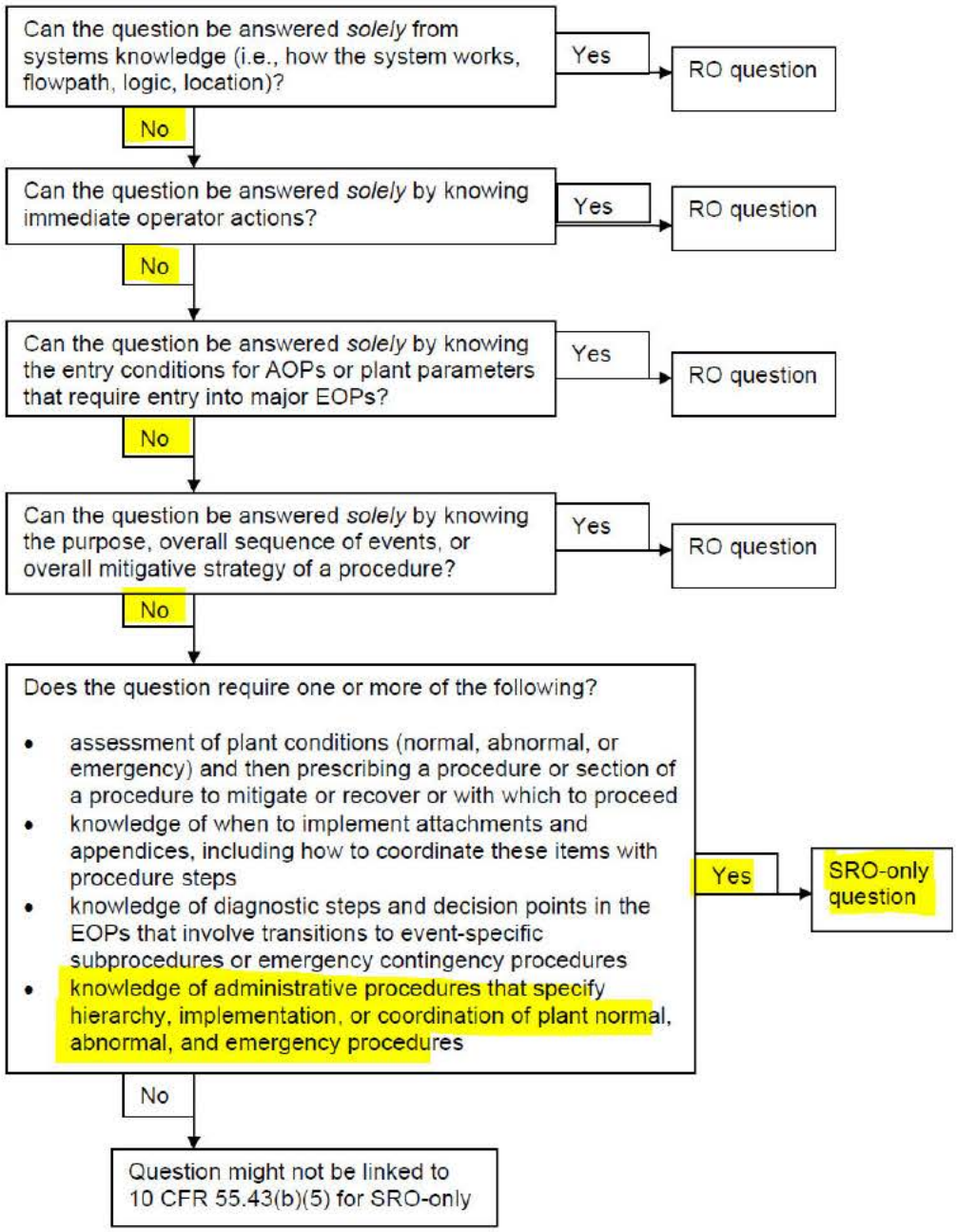
K/A Match: The K/A is matched because the question is asking the SRO to verify the status of the RCS leak on the CVCS system and determine whether or not this event is reportable to the States.

Additional Reference: S2.OP-AB.RC-0001 bases

At Step 3.24, the CRS/SM is notified to refer to the Event Classification Guide and to Technical Specifications. RCS Leakage definition for ECG classification purposes is "Any leakage that is unisolable should be considered Reactor Coolant System (RCS) Leakage when reviewing the Salem ECG for classification assessment". Leakage from Chemical Volume and Control System (CVCS) letdown piping, or components such as the 2CV6 relief valve or vent and drain valves associated with CVCS demineralizer vessels, is NOT considered Reactor Coolant System (RCS) Leakage. The CVCS system piping and components are capable of being isolated from the RCS piping and are classified in the Salem UFSAR as auxiliary to the reactor coolant system pressure boundary. In addition, Industry OE shows several plants have misclassified leakage from the CVCS as RCS Leakage and later retracted the TS Action Statement entry and Event notification. As a result, the entry into the Salem TS Action Statement for Identified Leakage is NOT required due to the source of the leakage being part of an isolable portion of the CVCS (non-RCS Leakage) and NOT from the RCS as defined in Technical Specifications.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



98 **14-086** **Points: 1.00**

Given:

- Salem Unit 2 is at 100% power
- A release of 21 CVCS Monitor Tank needs to be performed
- 2R18, Radioactive Liquid Effluent Monitor, was declared inoperable 3 days ago

Which of the following completes both statements?

1. Before performing the liquid radioactive waste release, the __(1)__authorizes the discharge.
2. Based on the above condition, the ODCM requires that _____(2)_____.

Note: ODCM = Offsite Dose Calculation Manual

A. (1) Shift Manager/Control Room Supervisor
(2) at least two independent samples of the monitor tank are analyzed prior to the release

B. (1) Shift Manager/Control Room Supervisor
(2) continuous surveys of the discharge piping are performed during the release

C. (1) Radiation Protection Manager
(2) continuous surveys of the discharge piping are performed during the release

D. (1) Radiation Protection Manager
(2) at least two independent samples of the monitor tank are analyzed prior to the release

Answer: A

Explanation / Justification:

A. Correct. Per the ODCM 3.3.8 Radioactive Liquid Effluent Monitoring Instrumentation, with 2R18 inoperable action 26 applies which requires at least two independent samples are analyzed and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving. Per S2.OP-SO.WL-0001, step 2.5 the SM/CRS authorizes the release.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because surveys of piping are performed but for different reason. Incorrect in that the ODCM requires two independent samples analyzed prior to the release which is verified by Chemistry.

C. Incorrect. Part 1 is incorrect. Plausible because the SRO may confuse the responsibility of a radioactive liquid release with a Rad Pro function. Part 2 is incorrect. Plausible because surveys of piping are performed but for different reason. Incorrect in that the ODCM requires two independent samples analyzed prior to the release which is verified by Chemistry.

D. Incorrect. Part 1 is incorrect. Plausible because the SRO may confuse the responsibility of a radioactive liquid release with a Rad Pro function. Part 2 is correct.

Technical References:	S2.OP-SO.WL-0001 (R32), ODCM 3.3.8 (R29)
Proposed References to be provided:	None
Learning Objective:	NOS05TECHSPEC-12, Objective 14. Describe the general component and parameter categories that are addressed by Technical Specification Sections 3/4.1 through 3/4.12Question Source: Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.13 / 43.4 / 45.10
Level:	SRO
Category/System:	14-Radiation Control
K/A #:	G2.3.6 Ability to approve release permits.
Importance:	3.8
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO requires knowledge of authorizing releases of liquid radioactive waste and additional requirements to perform the release if the radiation monitor is inoperable.

SRO Match Justification:

d. Radiation Hazards that May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO-only examination items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria or regulatory limits (or both)
- process for authorizing emergency exposure limits

99 **15-044** **Points: 1.00**

Given:

- The STA is monitoring Critical Safety Function Status trees

Which of the following completes the statement?

If Containment Radiation Monitors __(1)__ exceeds 2 R/HR, the CRS __(2)__ 2-EOP-FRCE-3, Response to High Containment Radiation Level.

A. (1) R10A/B
(2) must GO TO

B. (1) R44A/B
(2) must GO TO

C. (1) R44A/B
(2) may elect to GO TO

D. (1) R10A/B
(2) may elect to GO TO

Answer: C

Explanation / Justification:

A. Incorrect. Part 1 and 2 are incorrect.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because if the SRO believes a Red or Purple path exists, then entry into the FRP is required. Incorrect in that a yellow path exist and entry is optional.

C. Correct. Per 2-EOP-CFST-1, for containment environment, the R44A and R44B radiation monitors are used to determine whether entry into 2-EOP-FRCE-3 is required if radiation levels are above 2R/HR. FRCE-3 is a yellow path procedure, not RO knowledge, and the SRO must apply the rules of usage for FRPs to determine how the procedure is performed. Yellow path procedures are optional.

D. Incorrect. Part 1 is incorrect. Plausible because the R10A/B radiation monitors are located inside containment. Incorrect in that the R44A/B are the high range radiation monitors used in the CFSTs. Part 2 is correct.

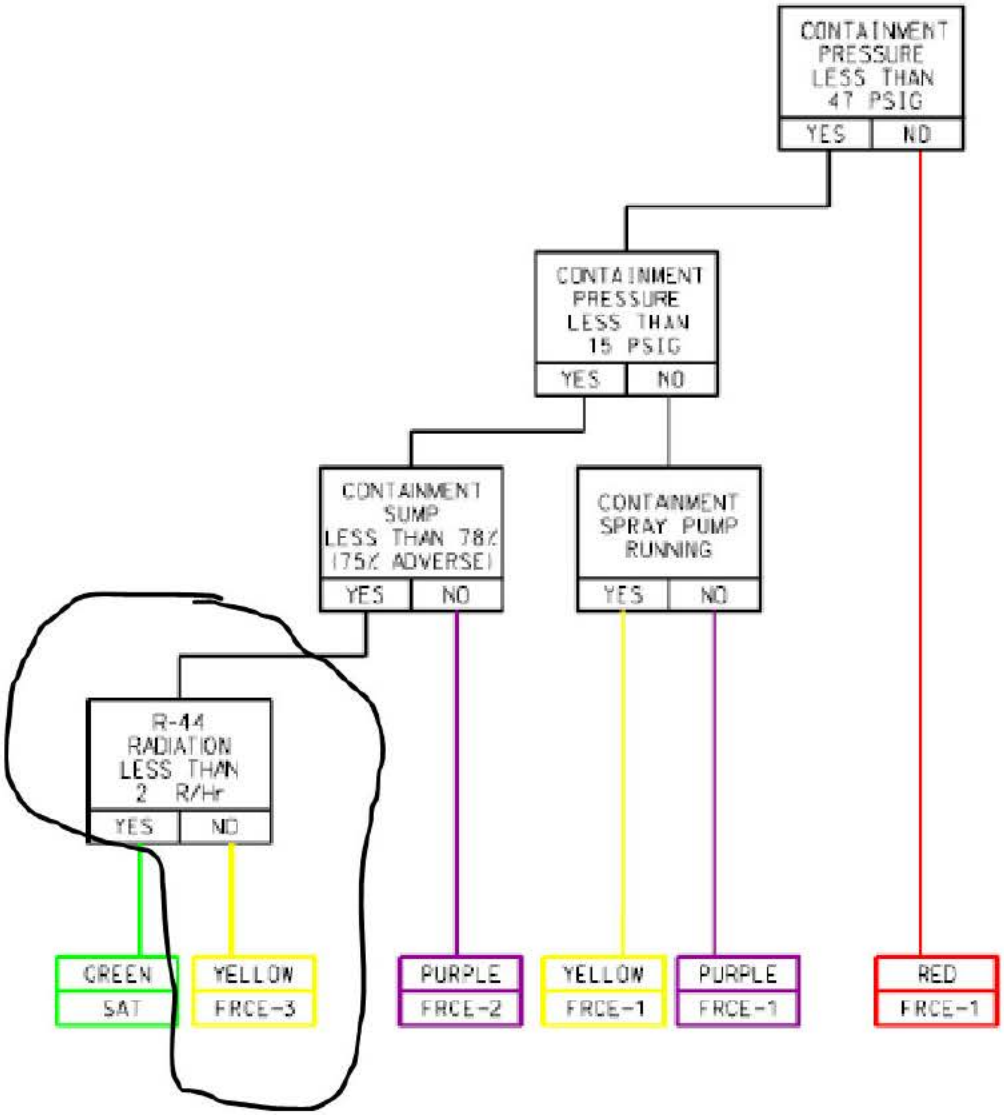
Technical References:	2-EOP-CFST-1 (R41), OP-SA-108-101-2002 (R0)
Proposed References to be provided:	None
Learning Objective:	NOS05TRP001-09, Objective 10, Given EOP-CFST-1 and a set of plant conditions, determine whether Functional Restoration Procedures (FRPs) should be implemented and/or terminated, in accordance with OP-SA-108-101-2002 and EOP-CFST-1.
Question Source:	Bank – Diablo Canyon 2018 NRC SRO85
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.7 / 45.7 / 45.8
Level:	SRO
Category/System:	15-Emergency Procedures/Plan
K/A #:	G2.4.2 Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions.
Importance:	4.6
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the question is asking the SRO which component is used to determine entry into a FRP procedure and based on the status of the trees, determine whether entry is required or optional.

Additional References: 2-EOP-CFST-1

FIGURE 5
CONTAINMENT ENVIRONMENT STATUS TREE

EOP-CFST-1



Additional Reference:

4.2.14. If a FRG is being performed, and a higher priority RED or PURPLE path comes in, the current FRG should be suspended and a transition made to the higher priority FRG unless stated otherwise in the applicable guideline. After the new FRG has been completed and guidance is provided to "return to procedure and step in effect," the operator should go back to the previous FRG which had been implemented, and was therefore the procedure in effect.

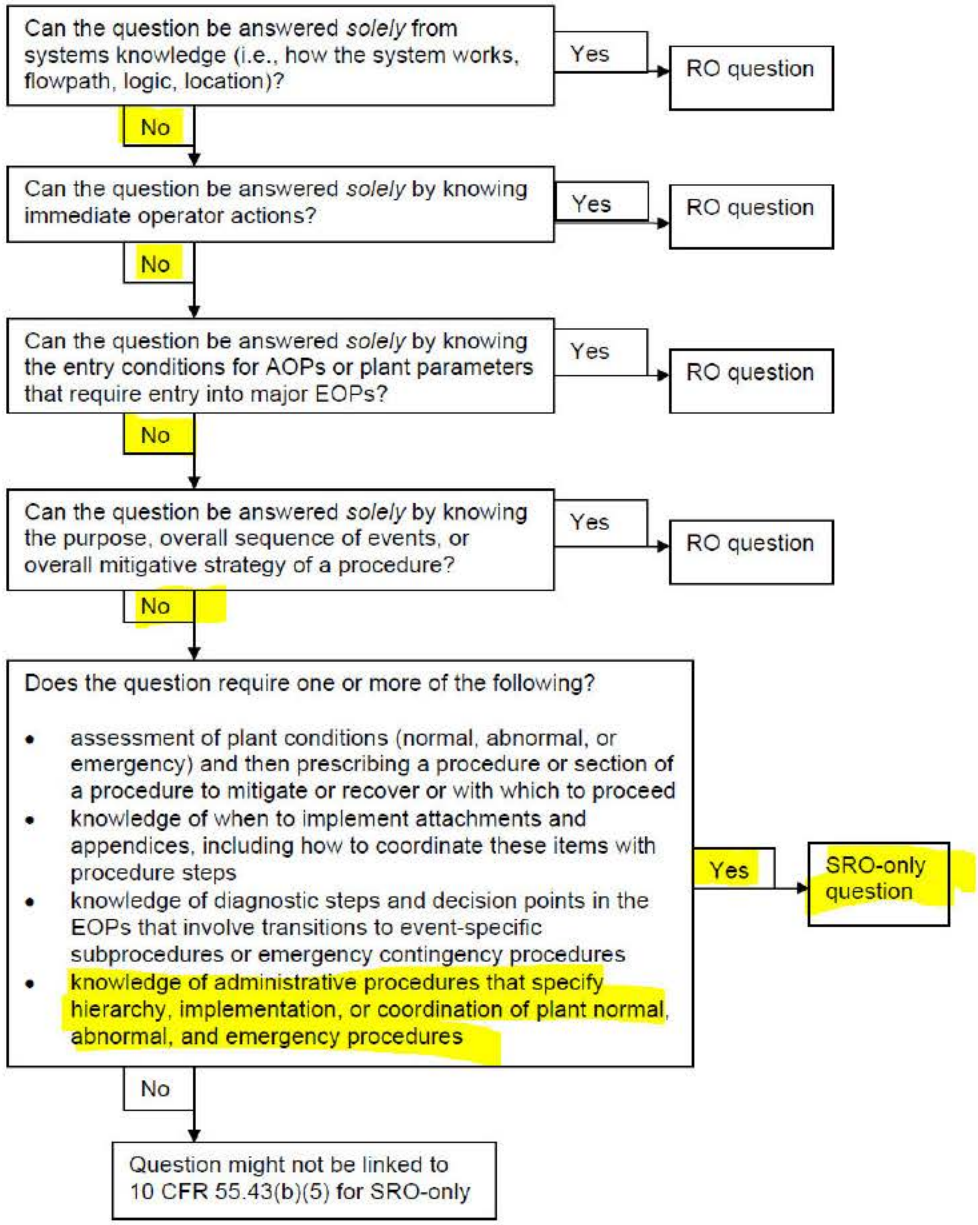
4.2.15. Status Tree monitoring should be continuous if any PURPLE or RED condition is found to exist. If a continuous RED or PURPLE priority is indicated, the performance of the appropriate CSF should be completed based on the priority rules of usage as discussed above. If no condition more serious than YELLOW is encountered, monitoring frequency may be reduced to 15 minutes, unless some significant change in plant status occurs.

4.2.16. A YELLOW terminus does not require immediate operator attention. Frequently it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the YELLOW status might provide an early indication of a developing RED or PURPLE condition. Following FRG implementation, a YELLOW might indicate a residual off-normal condition. The operator is allowed to decide whether or not to implement any YELLOW condition FRP.

4.2.17. Implementation of a yellow path function restoration procedure is based on operator judgement when it is determined that adequate time exists to implement it. In other words, the operator does not have to implement a yellow path procedure if a judgement has been made that it is inappropriate based on available time or current plant state; and, if an event of higher priority is in progress, the operator should attend to the more important matters prior to implementing a yellow path function restoration procedure.

SRO Match Justification:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



100 **15-045** **Points: 1.00**

Redacted due to security-related information

Given:

- A truck accident has rendered BOTH Salem Fire Pumps INOPERABLE
- The crew is implementing S2.OP-AB.FP-0001, Fire Protection System Malfunction

In accordance with S2.OP-AB.FP-0001, which of the following completes both statements?

1. What action will the CRS direct?

2. If the action cannot be established within 24 hours, then the unit must be in _____ within the next 6 hours?

A. (1) Open the backup supply cross-tie valves from Hope Creek.
(2) HOT STANDBY

B. (1) Open the backup supply cross-tie valves from Hope Creek.
(2) HOT SHUTDOWN

C. (1) [redacted]
(2) HOT SHUTDOWN

D. (1) [redacted]
(2) HOT STANDBY

Redacted due to security-related information

Redacted due to security-related information

Answer: A

Explanation / Justification:

A. Correct. Per S2.OP-AB.FIRE-0002, step 3.4 if at least one fire pump and one fire pump water supply is NOT available, then step 3.5 will direct opening the backup fire suppression water supply cross-tie valves from Hope Creek. If this cannot be established within 24 hours (step 3.6) then a unit shutdown is required and Hot Standby within the next 6 hours, Hot Shutdown within the following 6 hours, and Cold Shutdown within the subsequent 24 hours.

B. Incorrect. Part 1 is correct. Part 2 is incorrect. Plausible because Hot Shutdown is a required MODE, but in the next 6 hours after reaching Hot Standby in 6 hours.

C. Incorrect. Part 1 is incorrect. Plausible because [redacted], but not directed by procedure. Part 2 is incorrect. Plausible because Hot Shutdown is a required MODE, but in the next 6 hours after reaching Hot Standby in 6 hours.

D. Incorrect. Part 1 is incorrect. Plausible because [redacted] but not directed by procedure. Part 2 is correct.

Technical References:	S2.OP-AB.FIRE-0002 (R2)
Proposed References to be provided:	None
Learning Objective:	NOS05ABFP01-06, Objective 2. Describe, in general terms, the actions taken in S2.OP-AB.FP-0001 and the bases for the actions.
Question Source:	Bank - Salem
Question Cognitive Level:	Fundamental/Memory
10CFR Part 55 Content:	41.10 / 43.5 / 45.13
Level:	SRO
Category/System:	15-Emergency Procedures/Plan
K/A #:	G2.4.25 Knowledge of fire protection procedures.
Importance:	3.7
Tier/Group	Generic Knowledge and Abilities-Tier 3

K/A Match: The K/A is matched because the SRO is being asked the required action for when both Salem Fire Pumps are inoperable and the plant impact if the cross-tie cannot be opened within 24 hours.

Redacted due to security-related information

Additional Reference:

S2.OP-AB.FP-0001(Q)

___ 3.5 **NOTIFY** Site Protection to establish a Backup Fire Suppression Water System supply from Hope Creek, **OPEN** cross tie valves:

- 1PP30, FP HDR TO HOPE CREEK IV
- 0-KC-V115, HOPE CREEK/SALEM CROSSCONNECT VALVE

3.6 Can a backup Fire Suppression Water System supply be established within 24 hours?

___ NO ___ YES ——> **GO TO** Step 3.9 _____ Time

↓

V

___ 3.7 **NOTIFY** Operations Management of the unavailability of BOTH Fire Protection Water Systems.

___ 3.8 **INITIATE** Unit Shutdown IAW the appropriate IOP as follows: **[C0657]**

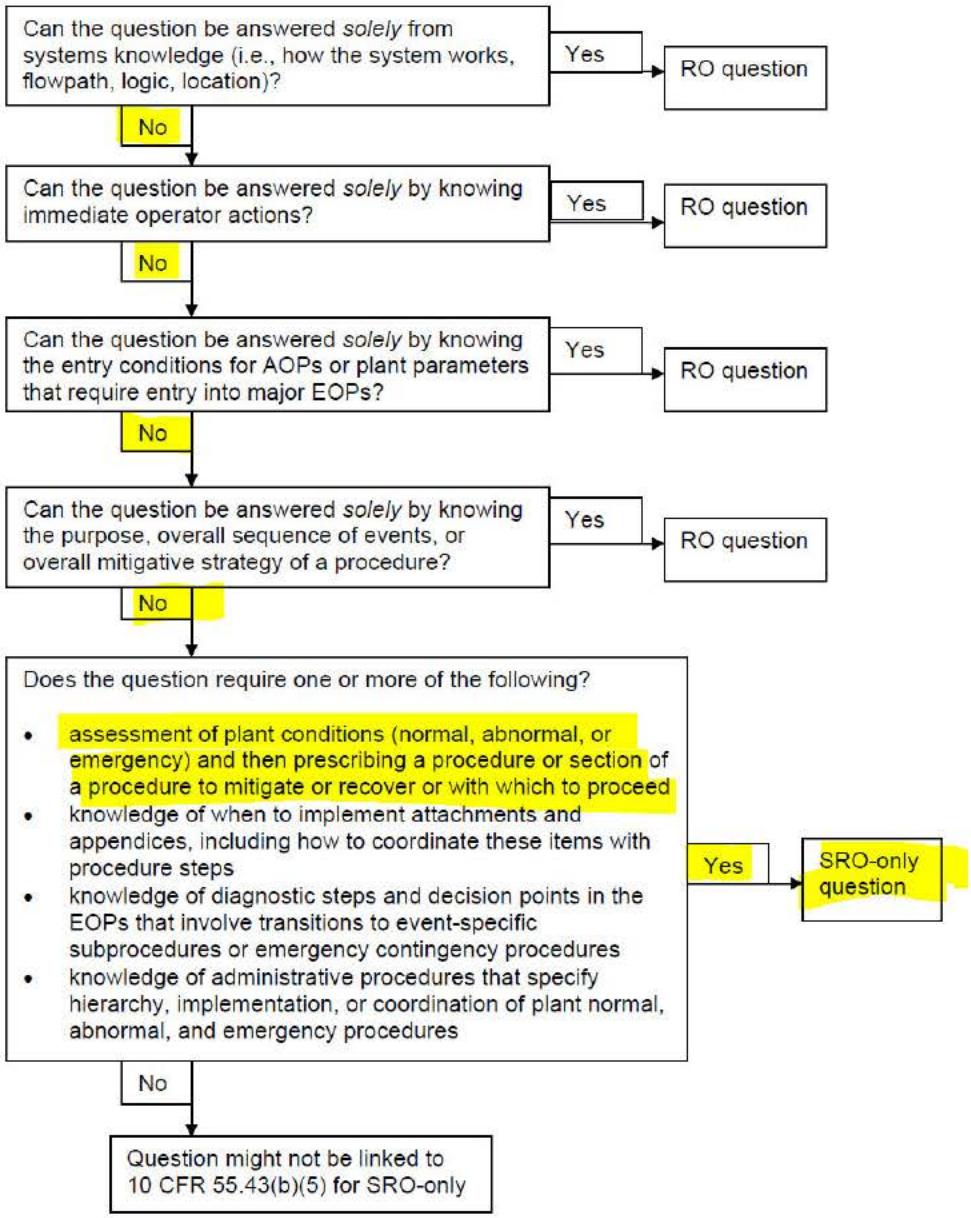
___ A. Be in at least HOT STANDBY within the next 6 hours.

___ B. Be in at least HOT SHUTDOWN within the following 6 hours.

___ C. Be in at least COLD SHUTDOWN within the subsequent 24 hours.

SRO Match Justification

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



SIMULATOR
EXAMINATION SCENARIO GUIDE

SCENARIO TITLE: NRC-1 [AB.COND-1, AB.NIS-1, AB.RC-1, TRIP-1, LOCA-1, LOCA-3]

SCENARIO NUMBER: 21-01 NRC ESG-1

EFFECTIVE DATE: See Approval Dates

EXPECTED DURATION: 80 minutes

REVISION NUMBER: 06

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **STA**

[ ] **OTHER** _____

Revision Summary:

- Rev 05 Incorporated NRC comments following phone call on 10-2-21 to delete SG pressure channel failure and modify RCS leak to 20 gpm to allow observing actions in AB.RC-1 and exercise Tech Specs.
- Rev 6 Modified scenario guide for use in Next ILOT exam 21-01

PREPARED BY:	K. Hantho **Lead Regulatory Exam Author**	9-15-22 **Date**
APPROVED BY:	M. Wadusky **Operations Training Manager or designee**	**Date**
APPROVED BY:	M. Winkelspecht **Facility Representative**	**Date**

SCAN OF SIGNED SCENARIO COVER SHEET

Form 3.3-1 Scenario Outline

Facility:	Salem	Scenario #:	1
Scenario Source:	20-01 ESG-1	Op. Test #:	21-01
Examiners:		Applicants/ Operators:	

Initial Conditions: Salem Unit 2 is at 100% power BOL

Turnover: The crew is directed to reduce power to 89% power at 10% per hour IAW S2.OP-IO.ZZ-0004 using boration, control rods and turbine load control in preparation for Main Turbine Valve testing.

Critical Tasks:

1. Close one Phase A isolation valve before transition out of TRIP-1
2. Transfer to Cold Leg Recirculation before receiving RWST Lo-Lo Alarm or ECCS pump cavitation

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	ATC(R) BOP(R) CRS(N)	Load reduction to 89% at 10%/hr IAW IOP-4 for turbine valve testing
2	VC0087	BOP (C,MC) CRS (C)	24 Vacuum Pump trips
3	NI0193A	ATC(I,MC,R) CRS(I,TS)	2N41 fails HIGH
4	RC0002	ATC (C) CRS (C,TS)	RCS leak (20 gpm) inside containment
5	RC0002 RC0001A	ALL (M)	RCS leak worsens to 350 gpm/ Large Break LOCA **(CT-3)**
6	RP0108	ATC (I,MC) CRS (I)	Automatic SI fails to actuate **(CT-1)**
7	RP318M1	ALL(I,MC)	#2 ECAC fails to start.
8	VL0045 VL0046	ATC(C,MC) CRS(C)	Two Phase A valves fail to close **(CT-2)**
		ABs	AB.COND-1 → AB.NIS-1 → AB.RC-1
		EOPs	TRIP-1 → LOCA-1 → LOCA-3

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

I. OBJECTIVES

A. Given the order, perform a turbine load reduction in accordance with station procedures.

B. Given indications of a loss of vacuum, perform actions as the nuclear control operator to RESPOND to the malfunction, IAW approved station procedures.

C. Given indications of a loss of vacuum, DIRECT the response to the malfunction IAW approved station procedures.

D. Given indications of a Power range nuclear instrument failure, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with approved station procedures.

E. Given indications of a Power range nuclear instrument failure, DIRECT the response to the malfunction in accordance with approved station procedures.

F. Given an RCS leak at power, respond to the event IAW approved station procedures.

G. Given the indications of a reactor coolant system (RCS) malfunction or leak, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with the approved station procedures.

H. Given the indications of a reactor coolant system (RCS) malfunction or leak, DIRECT the response to the malfunction in accordance with the approved station procedures.

I. Given the order or indications of a reactor trip, perform actions as the nuclear control operator to RESPOND to the reactor trip in accordance with the approved station procedures.

J. Given indication of a reactor trip, DIRECT the response to the reactor trip in accordance with the approved station procedures.

K. Given the order or indications of a safety injection perform actions as the nuclear control operator to RESPOND to the safety injection in accordance with the approved station procedures.

L. Given indication of a safety injection DIRECT the response to the safety injection in accordance with the approved station procedures.

M. Given a safety injection has occurred and equipment has failed to start, START equipment that has failed to automatically start in accordance with station procedures.

N. Given the order or indications of a loss of coolant accident (LOCA), complete actions as the nuclear control operator to PERFORM the immediate response to the LOCA in accordance with the approved station procedures.

O. Given indication of a loss of coolant accident (LOCA), DIRECT the immediate response to the LOCA in accordance with the approved station procedures.

P. Given the order or a loss of coolant accident (LOCA) and plant conditions to support cold leg recirculation, perform actions as the nuclear control operator to TRANSFER to cold leg recirculation in accordance with the approved station procedures.

Q. Given a loss of coolant accident (LOCA) and plant conditions to support cold leg recirculation, DIRECT actions to transfer to cold leg recirculation in accordance with the approved station procedures.

R. During performance of emergency operating procedures, monitor the critical safety function status trees in accordance the EOP in effect.

II. MAJOR EVENTS

1. Planned Load Reduction to 89% at 10%/hour
2. 24 Vacuum Pump trips
3. 2N41 Fails High (TS)
4. RCS Leak (20 gpm) (TS)
5. LBLOCA
6. Auto SI fails to actuate
7. #2 ECAC Fails to Start on SEC Signal
8. Two Phase A valves 2CV116, and 284 fail to close

III. SCENARIO SUMMARY

1. The crew assumes the watch at 100% power, EOL.
2. After assuming the watch, the crew will brief a planned load reduction to 89% at 10%/hour IAW **S2.OP-IO.ZZ-0004**, Power Operation, in preparation for Main Turbine Valve testing next shift.
3. After the power reduction has commenced, 24 condenser vacuum pump will trip. The crew will recognize that condenser backpressure is slowly rising and enter **S2.OP-AB.COND-0001**, Loss of Condenser Vacuum. The crew will start all available vacuum pumps and stabilize condenser back pressure. The 25AR25 will fail to open and the operator will manually open from control room.
4. After the crew addresses the vacuum pump trip, 2N41 Power Range channel will fail high. The operator will respond by placing rod control in Manual. The crew will enter **S2.OP-AB.NIS-0001,** NIS Malfunction**,** and perform actions to remove channel from service. The CRS will evaluate Tech Specs.
5. Once actions for 2N41 Power Range channel are addressed, a RCS leak (20 gpm) will occur resulting in lowering PZR level and rising counts on the 2R11A containment radiation monitor. The CRS will enter **S2.OP-AB.RC-0001, RCS Leak**, and adjust Charging flow to stabilize PZR level. The CRS will evaluate TS and enter TS 3.4.7.2.b, Action b. Once TS has been evaluated, the RCS leak will worsen to 350 gpm. The crew will take CAS actions in S2.OP-AB.RC-0001 and manually trip the reactor, at which time the RCS leak will escalate into a Large Break LOCA.
6. Immediately following the Rx Trip a malfunction will prevent Auto Safety Injection from actuating. The crew will respond by manually initiating SI. The crew will enter **EOP-TRIP-1,** Reactor Trip or Safety Injection.
7. While in EOP-TRIP-1, the crew will recognize that #2 ECAC failed to start on SEC signal. The crew will block and reset 2C SEC, then manually start #2 ECAC, additionally two Phase A valves will fail to close and the crew will take action to close at least one valve in the penetration. [**Critical Task #1**].
8. The crew will perform diagnostics in EOP-TRIP-1 and eventually transition to **EOP-LOCA-1**, Loss of Reactor Coolant and **EOP-LOCA-3**, Transfer to Cold leg recirculation. While in LOCA-3

they will establish cold leg recirculation. [**Critical Task #2 – this CT includes time critical operator actions**].

9. The scenario can be terminated when the crew reduces ECCS to one train, stops all ECCS pumps taking suction from the RWST if the Lo-Lo level alarm is received, or by direction from Lead Examiner.

IV. INITIAL CONDITIONS

____ **IC-240**

PREP FOR TRAINING (i.e. computer setpoints, procedures, bezel covers ,tagged equipment)

Initial	Description
____ 1	2VC1and 2VC4 C/T
____ 2	RCPs (SELF CHECK)
____ 3	RTBs (SELF CHECK)
____ 4	21-24MS167s (SELF CHECK)
____ 5	500 KV SWYD (SELF CHECK)
____ 6	21 and 22 SGFP Trip (SELF CHECK)
____ 7	**Suggested Protected Equipment:** ▪ **none**
____ 8	Complete Attachment 2 “Simulator Ready-for-Training/Examination Checklist.”

Note: Tables with blue headings may be populated by external program, do not change column name without consulting Simulator Support group

EVENT TRIGGERS:

Initial	ET #	Description	
	1	EVENT ACTION: COMMAND: PURPOSE:	kck05anz //25AR25 25 VACUUM PUMP-CONDENSER DMF VL0440 <update as needed>
	2	EVENT ACTION: COMMAND: PURPOSE:	MONP254 < 10. //CONT ROD BANK C < 10 (RX TRIP) <update as needed>
	3	EVENT ACTION: COMMAND: PURPOSE:	kb116tcb //2CV116 SEAL W TO VCT CLOSE DMF VL0045 <update as needed>
	4	EVENT ACTION: COMMAND: PURPOSE:	kb117tcb //2CV284 SEAL W TO VCT CLOSE DMF VL0046 <update as needed>
	5	EVENT ACTION: COMMAND: PURPOSE:	KCK02PBZ //22 VACUUM PUMP-START <update as needed>

MALFUNCTIONS:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
_____ 01	VC0087C TRIP OF VACUUM PUMP 24	N/A	N/A	N/A	RT-1	
_____ 02	VC0087A TRIP OF VACUUM PUMP 22	00:00:15	N/A	N/A	ET-5	
_____ 03	VL0440 25AR25 Fails to Position (0-100%)	N/A	N/A	N/A	N/A	0
_____ 04	NI0193A PR CH N41 FAILS HI/LO	N/A	N/A	N/A	RT-2	150
_____ 05	RC0002 RCS LEAK INTO CONTAINMENT (equiv to 0-4 inches)	N/A	N/A	N/A	RT-3	20
_____ 06	RC0001A RCS RUPTURE OF RC LOOP 21	N/A	N/A	N/A	ET-2	
_____ 07	RP0108 FAILURE OF AUTOMATIC SI	N/A	N/A	N/A	N/A	
_____ 08	RP318M1 EMER Contrl Air Compressor Fails to Start on SEC	N/A	N/A	N/A	N/A	
_____ 09	VL0045 2CV116 Fails to Position (0-100%)	N/A	N/A	N/A	N/A	100
_____ 10	VL0046 2CV284 Fails to Position (0-100%)	N/A	N/A	N/A	N/A	100

REMOTES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition

OVERRIDES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity

OTHER CONDITIONS:

	Description
____ 1.	None

V. SEQUENCE OF EVENTS

A. State shift job assignments.

B. Hold a shift briefing, detailing instruction to the shift: (provide crew members a copy of the shift turnover sheet)

C. Inform the crew “The simulator is running. You may commence panel walkdowns at this time. SM please inform me when your crew is ready to assume the shift”.

D. Allow sufficient time for panel walk-downs. When informed by the SM that the crew is ready to assume the shift, ensure the simulator is cleared of unauthorized personnel.

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
1. **Power Reduction to 89%**			
	CRS briefs crew on power reduction to 89% at 10%/hour IAW S2.OP-IO.ZZ-0004, Section 4.3 Power Reduction.		
	RO briefs boration plan for load reduction.		
	PO briefs turbine load control plan.		
	RO initiates boration IAW S2.OP-SO.CVC-0006.		
	PO initiates turbine load reduction IAW S2.OP-SO.TRB-0002, Turbine Generator Shutdown Operation.		
	RO monitors Tavg and control rods for proper response.		
Proceed to next event by direction from Lead Examiner.			
Examiner's Note: The crew may at any time stop the load reduction to respond to the next events.			
2. 24 Vacuum Pump Trips:			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
Simulator Operator: At direction from Lead Evaluator insert **RT-1**. This will trip 24 vacuum pump: **VC0087C, Trip of Vacuum Pump 24**			
	PO reports unexpected trip of 24 Vacuum Pump.		
	CRS enters S2.OP-AB.COND-0001, Loss of Condenser Vacuum.		
	PO initiates Attachment 1 CAS.		
	CRS dispatches operator to perform Attachment 2, Loss of Vacuum Local Checks.		
Role Play: When dispatched to perform Attachment 2, THEN after 2-5 minutes report the following: ***I've completed Attachment 2 and have found no abnormal conditions except for the vacuum pump breaker being tripped open* on 22 and 24 vacuum pumps.**			
	PO reports 24 Vacuum pump is stopped and the 24AR25 is closed.		
	PO reports condenser backpressure is rising rapidly.		
	PO reports rising backpressure is not attributed to CW System malfunctions.		
	PO starts all available vacuum pumps.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
22 Vacuum Pump trips following start:			
Simulator Operator: Ensure **ET-2** is TRUE IF 22 Vacuum pump is started. This will cause the pump to trip following start.			
	IF 22 Vacuum Pump is started, the PO reports that the pump started then tripped seconds later.		
25AR25 fails to open following start of 25 Vacuum Pump:			
	PO reports that 25 Vacuum Pump started but the 25AR25 failed to open.		
Role Play: IF requested to check local indications at the Vacuum Pump, THEN report that ***the local gauge reading is within the normal band and all other parameters are within their normal bands.***			
	PO manually opens 25AR25.		
	PO reports that condenser back pressure is improving.		
Proceed to next event at direction from Lead Examiner.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
3. 2N41 Power Range Channel fails high **Simulator Operator:** Enter **RT-2** on direction of Lead Evaluator. **MALF: NI0193A PR CH41 Fails High Severity: 150**			
	RO reports rods stepping in and no runback in progress		
	RO places rod control in Manual		
	CRS enters S2.OP-AB.NIS-0001, NIS Malfunction.		
Evaluator's Note: The crew may enter S2.OP-AB.ROD-0003 which will transition you to AB.NIS-0001			
	RO reports rod control is in Manual		
	PO reports no turbine load change in progess		
	RO reports that 2N41 channel is failed high		
	RO reports the following OHA alarms are in: ▪ E-15 PR HI RNG FLUX HI ▪ E-31 PR OVRPWR ROD STOP ▪ E-39 PR CH DEV ▪ E-47 PR NEUT FLUX RATE HI		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	CRS directs PO to remove PR channel from service IAW S2.OP-SO.RPS-0001, NI Channel Trip/Restoration		
Evaluator's Note: At the direction of the Lead Evaluator, performing removing of 2N41 from service is optional. Performing these steps takes about **10 minutes**.			
Placing 2N41 in Tripped Condition IAW S2.OP-SO.RPS-0001, Steps 4.1.1 to 4.1.5:			
	PO ensures that tripping of associated bistable(s) will NOT result in an ESF OR RPS actuation		
	PO ensures 2N41 Channel is NOT selected on NIS Recorder 2NR45		
	PO ensures Rod Control is in Manual		
	CRS enters T/S 3.3.1.1, Actions 2 and 6 (72 hours) Reactor Trip System Instrumentation for 2N41		
	At NI Rack No. 81, PO performs the following:		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	PO places DETECTOR CURRENT COMPARATOR, UPPER SECTION, switch in PRN41 position AND ENSURE the following: ___ CHANNEL DEFEAT lamp illuminates. ___OHA E-38, UPPER SECT DEV ABV 50% PWR, clears. Note: E-38 was not in.		
	PO places DETECTOR CURRENT COMPARATOR, LOWER SECTION, switch in PRN41 position AND ENSURE the following: ___CHANNEL DEFEAT lamp illuminates. ___OHA E-46, LOWER SECT DEV ABV 50% PWR, clears. Note E-46 was not in.		
	PO places POWER MISMATCH BYPASS switch in BYPASS PR N41. (Defeats input to Rod Control)		
	PO places ROD STOP BYPASS switch in BYPASS PR N41AND ENSURE the following: ___ 2RP4 - OVER POWER ROD STOP MANUAL BYPASS, CH I is illuminated. ___ OHA E-31, PR OVERPWR ROD STOP, is clear		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	PO places COMPARATOR CHANNEL DEFEAT switch in N41 AND ENSURE the following: ___COMPARATOR DEFEAT lamp is illuminated. ___ OHA E-39, PR CH DEV, is clear		
Evaluator's Note: After Step 4.1.5 is complete Maintenance support is needed to complete the rest of the procedure for tripping bistables. Rod control can be placed in Auto at this time if the CRS directs. Leaving Rods in Manual has no affect to the outcome of the scenario.			
	CRS directs Maintenance to remove 2N41 channel from service IAW S2.OP-SO.RPS-0001.		
	RO withdraws rods as required to maintain band prescribed.		
	CRS may elect to restore rod control to Auto or leave in Manual until the bistables are tripped.		
Proceed to next event after rods are withdrawn to ARO or by direction from Lead Evaluator.			
4. RCS Leak (20 gpm)			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
Simulator Operator: Insert **RT-3** by direction from Lead Examiner. **RC0002, RCS Leakage** **Value = 20** **Ramp = None**			
	RO reports counts on the 2R11A containment radiation monitor is rising.		
	RO reports PZR level is lowering and/or RCS pressure is lowering.		
	RO reports RCS leak inside containment.		
	RO determines RCS leak rate.		
Examiners Note: The CRS can also enter S2.OP-AB.RAD-0001, Abnormal Radiation, in response to OHA A-6. If so, the CRS can direct the PO to perform actions of AB.RAD-1 while the CRS directs actions in AB.RC-1.			
	CRS enters S2.OP-AB.RC-0001, RCS Leakage.		
	PO initiates Attachment 1 CAS		
	RO reports RCS temperature is above 350 °F.		
	RO reports Unit is not in Mode 3.		
	RO reports PZR level is lowering.		
	RO swaps to a centrifugal charging pump in service.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	CRS directs RO to adjust charging flow to stabilze PZR level.		
	RO adjusts charging flow to stabilize PZR level and then determines RCS leak rate.		
TS evaluation #2:			
	CRS enters: • **TS 3.4.7.2.b (1 gpm unidentified leakage), action b (reduce leakage rate to within limits within 4 hours or be in Hot Standby)**		
Proceed on to next event after TS has been evaluated or by direction from the Lead Examiner			
5. RCS leak worsens to 350 gpm and Large Break LOCA (Major Transient):			
Simulator Operator: Modify **RT-3** to 350 at the direction of the Lead Examiner. **RC0002, RCS Leakage** **Value = 350** **Ramp = None**			
	RO reports PZR level is lowering rapidly.		
	RO attempts to maintain PZR level by adjusting charging flow.		
	RO reports RCS leak rate exceeds make up capability.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	CRS implements the Attachment 1 CAS action and briefs the RO to Trip the Reactor and actuate Safety Injection.		
	RO trips the reactor and actuates SI.		
Simulator Operator: Ensure **ET-1** is TRUE following Rx Trip. This will insert LBLOCA malfunction. **RC0001A, RCS rupture on RC loop 21**			
6. Auto SI fails to actuate on both trains:			
	RO may report SI failed to Auto actuate (depending on where RCS pressure is at during time SI is manually actuated)		
	RO manually actuates SI on one Train and verifies the other train actuated.		
	RO continues Immediate Actions of TRIP-1: ▪ Reports the Main Turbine is tripped and backs up Main Turbine trip. ▪ Reports all 4KV vital busses energized. ▪ Reports SI has been initiated.		
	CRS and RO verify Immediate Actions complete.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
7. # 2 ECAC fails to start on SEC signal	**CRS directs initiation of the following EOP-TRIP-1 CAS actions when RCS pressure meets the criteria:** • **Stops RCPs (1240(1350 adverse) psig)** • **Close Charging Mini-flows (1500 psig)**		
	PO reports ALL Vital Buses are energized.		
	PO reports SEC loading for 2B and 2C vital bus is NOT complete.		
	PO reports # 2 ECAC failed to start.		
	PO blocks and resets 2C SEC.		
	RO Starts #2 ECAC.		
	PO throttles AFW flow to no less than 22E4lbm/hr while SG NR levels remain <9%. (15% adverse)		
	RO reports containment pressure has NOT remained <15 psig.		
	RO reports Phase B and Spray actuation Auto initiated.		
	RO reports 21 and 22 CS pumps are running.		
	RO reports MSLI Auto initiated.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	RO reports all valves groups in Table B are in the safeguards positions.		
	RO Stops all RCPs.		
	PO reports 2RP4 indicates High-High Containment Pressure.		
	RO reports both trains of MSLI has actuated.		
	PO reports all valves in Table D are closed.		
	RO reports 2 CCW pumps are running.		
	RO reports both CCW HXs are in Auto.		
	RO reports 2CC131 (Thermal Barrier Return) is closed.		
	RO reports Phase B has occurred and 2CC131 should not be opened.		
	CRS notifies WCC to monitor Spent Fuel Pool temperature and level.		
	RO reports RWST Lo Level Alarm has not actuated (this is a CAS step)		
	PO reports 2CV116 and 284 in Table E are NOT in the safeguards positions.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	Critical task #1 - RO closes the 2CV116 and 284 from the 2CC2 cosole pushbuttons		
Critical Task-1 (CT-11): Close at least one valve on each phase A penetration before transition out of EOP-TRIP-1, Reactor Trip or Safety Injection. **SAT_____ UNSAT_____**			
	RO reports 21 and 22 CA330s are closed.		
	RO reports CAV is in Accident Pressurize Mode.		
	RO reports CAV is not in Maintenance Mode Alignment		
	RO reports two switchgear fans and one switchgear exhaust fan are running		
	RO reports Charging, SI pump and RHR pump flows are consistent with RCS pressure.		
	PO maintains total AFW flow >22E4 lbm/hr until at least one SG NR level is > 9% (15% adverse), then maintains SG NR levels between 19% and 33%.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
Examiner's Note: Based on the pace of the crew, they may perform the following CAS steps sooner or later in EOP-TRIP-1.			
	RO closes charging pump mini flows IAW TRIP-1 CAS when RCS pressure is < 1500 psig with BIT flow established. [TRIP-1 CAS]		
	RO stops RCPs IAW TRIP-1 CAS when RCS pressure is <1240(1350 adverse) psig with ECCS flow established. [TRIP-1 CAS]		
	RO reports RCP status, and CRS determines whether a MSLI for temperature control is required.		
	RO reports PZR spray valves response are normal.		
	RO reports RCS pressure < 1240(1350 adverse) psig and ECCS flow established.		
Examiner's Note: SG pressure could be lowering slowly based on the break flow from the LBLOCA.			
	PO reports NO indications of a Faulted SG exist.		
	PO reports NO indications of a SGTR exist.		
	RO reports two or more channels in Table J are NOT indicating normal on the P250 computer **OR** reports Containment Pressure > 4 psig.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
Evaluator's Note: **IF** RWST low level (15.2 ft.) is reached during TRIP-1, **THEN** go to **page 27 for LOCA-3 steps**.			
EOP-LOCA-1 steps here:			
	CRS transitions to 2-EOP-LOCA-1, Loss of Reactor Coolant.		
Examiner's Note: During LOCA-1, the crew will reach the low RWST alarm (15.2 feet) and then transition to EOP-LOCA-3.			
Examiners Note: A CFST PURPLE Path on Thermal Shock may exist during LOCA-1, if so, the crew will enter and exit FRTS-1 with no actions performed.			
	RO reports all RCPs are stopped.		
	PO reports no indications of faulted SGs.		
	PO maintains total AFW flow >22E4 lbm/hr until at least one SG NR level is > 9% (15% adverse), then maintains SG NR levels between 19% and 33%.		
	RO resets Phase A.		
	RO opens 21 and 22 CA330s.		
	RO resets SG B/D Sample Isolation Bypass		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	RO opens 21 thru 24 SS94s (SG B/D Sample Valves)		
	PO reports ALL secondary radiation monitors in Table B are indicating Normal.		
	RO reports both PZR PORVs are Closed and Block valves are Open.		
	RO reports subcooling is NOT greater than 0 F.		
	RO reports both CS Pumps are running.		
	RO resets both trains of SI.		
	PO resets each SECs.		
	RO reports RCS pressure is not greater than 300 psig.		
	Crew stops unloaded EDG's		
	Crew determines LOCA 5 and 6 transition not required and consult with the TSC		
	Crew will now loop back around to Step 11 or transition to LOCA-3.		
FRTS-1 steps here:			
	CRS enters 2-EOP-FRTS-1, Response to Imminent Pressurized Thermal Shock Conditions.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	STA verifies procedure transition.		
	RO reports RCS pressure is < 420 psig and RHR flow is at least 300 gpm.		
	CRS returns to procedure in effect (LOCA-1).		
Record Time RWST Lo Level Alarms Actuated: **Time:_____:_____:_____**			
8. Transfer to Cold Leg Recirculation (EOP-LOCA-3)			
Evaluator's Note: A predetermined Critical Task which includes Time Critical Operator Actions			
	RO reports Containment Sump CH A (B) Level > 62% is illuminated.		
	RO depresses SUMP AUTO ARMED PB for 21 and 22 SJ44s.		
	PO removes lockouts from 2SJ67, 2SJ68, 2SJ69.		
	RO reports 21 and 22 SJ44s open.		
	RO reports both RHR pumps running.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	RO closes 2SJ69.		
Mark time 2SJ69 close PB depressed. _____:_____:_____ **≤ 3.7 mins after RWST lo level**			
	RO verifies or performs SI reset actions.		
	RO resets SECs and 230V MCCs..		
	RO stops 22 Containment Spray Pump.		
Mark time 22 CS pump stopped. _____:_____:_____ **≤ 5.5 mins after RWST Lo Level**			
	RO closes 21 and 22 RH19s.		
	RO stops 23 charging pump.		
	CRS goes to **Step 11** based on all Vital Buses energized.		
	RO reports 3 SW pumps running.		
	PO reports both CCHXs are in service OR being restored to service per EOP-APPX-1.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
	RO reports that 22 CS pump is NOT running.		
	RO reports that 21 and 22 CC16 are open.		
	RO performs the following: ▪ **Closes 2SJ67 and 2SJ68.** ▪ **Reports 2RH1 and 2RH2 are closed.**		
	RO opens 21 and 22 SJ45s.		
	RO reports 21 and 22 SJ113's are open.		
	RO reports 21 and 22 SI pumps and 21 and 22 Charging pumps are running.		
Mark time ECCS pumps verified in operation. **_____:_____:_____** **≤ 11.2 mins after RWST Lo Level**			
	PO removes lockout for 2SJ30		
	RO closes 2SJ30, 2SJ1, and 2SJ2		
	RO places 21RH29 and 22RH29 in MANUAL and reports both valves are closed.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments
Critical Task-2 (CT-36): Transfer to Cold Leg Recirculation before RWST Lo-Lo Level (1.2 feet) is reached or ECCS pump cavitation. This CT also includes accomplishing operator time critical actions as specified below: **Time Critical Operator Actions:** Transfer to Cold Leg Recirculation such that at least one train of ECCS is in operation in the recirculation mode within the following time frames: **1. From RWST lo level alarm to initiating closed on 2SJ69 - ≤3.7 minutes** **2. From RWST lo level alarm to one containment spray pump stop ≤5.5 minutes** **3. From RWST lo level alarm to switchover completion (includes restarting ECCS pumps if stopped on RWST lo-lo level)** **≤ 11.2 minutes.** **SAT_____ UNSAT_____**			
Terminate scenario when the crew initiates close or verifies closed for RH29s or at Lead Evaluator direction.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comments

VI. SCENARIO REFERENCES

A. Alarm Response Procedures (various)
B. Technical Specifications
C. Emergency Plan (ECG)
D. S2.OP-AB.COND-0001, Loss of Condenser Vacuum
E. S2.OP-AB.ROD-0003, Continuous Rod Motion
F. S2.OP-AB.NIS-0001,Nuclear Instrumentation System Malfunctions
G. S2.OP-AB.RC-0001, RCS Leak
H. 2-EOP-TRIP-1, Reactor Trip or Safety Injection
I. 2-EOP-LOCA-1, Loss of Reactor Coolant
J. 2-EOP-LOCA-3, Transfer to Cold leg recirculation

ATTACHMENT 1
UNIT TWO PLANT STATUS
TODAY

MODE: 1 POWER: 100% RCS BORON: 967 ppm MWe 1220

SHUTDOWN SAFETY SYSTEM STATUS (5, 6 & DEFUELED):

NA

REACTIVITY PARAMETERS:

- Rx Plan: To lower Rx power to 89% at 10%/hour add 120 gallons of boric acid along with inserting control rods to maintain Tavg on program. Reactor Engineering directs the crew to perform a boration first. Expect D bank at 193 steps when load reduction is complete.

MOST LIMITING LCO AND DATE/TIME OF EXPIRATION:

- None

EVOLUTIONS/PROCEDURES/SURVEILLANCES IN PROGRESS:

- Load reduction to 89% at 10%/hour in preparation for Main Turbine valve testing.

ABNORMAL PLANT CONFIGURATIONS:

- None

CONTROL ROOM:

- Hope Creek and Salem 1 are at 100% power.

PRIMARY:

- none

SECONDARY:

- Heating Steam is aligned to Unit 1
- Polisher is in service
- Blowdown is 35K per loop to 23 Condenser / Flashtank

RADWASTE:

- No discharges in progress

CIRCULATING WATER/SERVICE WATER:

ATTACHMENT 2

SIMULATOR READY-FOR-TRAINING CHECKLIST

____ 1. Verify simulator is in "TRAIN" Load

____ 2. Simulator is in RUN

____ 3. Overhead Annunciator Horns ON

____ 4. All required computer terminals in operation

____ 5. Simulator clocks synchronized

____ 6. All tagged equipment properly secured and documented

____ 7. TSAS Status Board up-to-date

____ 8. Shift manning sheet available

____ 9. Procedures in progress open and signed-off to proper step

____ 10. All OHA lamps operating (OHA Test) and burned out lamps replaced

____ 11. Required chart recorders advanced and ON (proper paper installed)

____ 12. All printers have adequate paper AND functional ribbon

13. Required procedures clean

____ 14. Multiple color procedure pens available

____ 15. Required keys available

____ 16. Simulator cleared of unauthorized material/personnel

____ 17. All charts advanced to clean traces and chart recorders are on.

____ 18. Rod step counters correct (channel check) and reset as necessary

____ 19. Exam security set for simulator

____ 20. Ensure a current RCS Leak Rate Worksheet is placed by Aux Alarm Typewriter

____ With Baseline Data filled out

____ 21. Shift logs available if required

____ 22. Recording Media available (if applicable)

____ 23. Ensure ECG classification is correct

____ 24. Reference verification performed with required documents available

____ 25. Verify phones disconnected from plant after drill.

____ 26. Verify EGC paperwork is marked "Training Use Only" and is current revision.

____ 27. Ensure sufficient copies of ECG paperwork are available.

ATTACHMENT 3
CRITICAL TASK METHODOLOGY

In reviewing each proposed CT, the examination team assesses the task to ensure, that it is essential to safety. A task is essential to safety if, in the judgment of the examination team, the improper performance or omission of this task by a licensee will result in direct adverse consequences or in significant degradation in the mitigative capability of the plant.
The examination team determines if an automatically actuated plant system would have been required to mitigate the consequences of an individual's incorrect performance. If incorrect performance of a task by an individual necessitates the crew taking compensatory action that would complicate the event mitigation strategy, the task is safety significant.

I. Examples of CTs involving essential safety actions include those for which operation or correct performance prevents...

- degradation of any barrier to fission product release
- degraded emergency core cooling system (ECCS) or emergency power capacity
- a violation of a safety limit
- a violation of the facility license condition
- incorrect reactivity control (such as failure to initiate Emergency Boration or Standby Liquid Control, or manually insert control rods)
- a significant reduction of safety margin beyond that irreparably introduced by the scenario

II. Examples of CTs involving essential safety actions include those for which a crew demonstrates the ability to...

- effectively direct or manipulate engineered safety feature (ESF) controls that would prevent any condition described in the previous paragraph.
- recognize a failure or an incorrect automatic actuation of an ESF system or component.
- take one or more actions that would prevent a challenge to plant safety.
- prevent inappropriate actions that create a challenge to plant safety (such as an unintentional Reactor Protection System (RPS) or ESF actuation.

ATTACHMENT 4
SIMULATOR SCENARIO REVIEW CHECKLIST

SCENARIO IDENTIFIER: 21-01 NRC Scenario #1 REVIEWER: K. Hantho

Initials		Qualitative Attributes
K	1.	The scenario has clearly stated objectives in the scenario.
K	2.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue crew into expected events.
K	3.	The scenario consists mostly of related events.
K	4.	Each event description consists of: • the point in the scenario when it is to be initiated • the malfunction(s) that are entered to initiate the event • the symptoms/cues that will be visible to the crew • the expected operator actions (by shift position) • the event termination point
K	5.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.
K	6.	The events are valid with regard to physics and thermodynamics.
K	7.	Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.
K	8.	The simulator modeling is not altered.
K	9.	All crew competencies can be evaluated.
K	10.	The scenario has been validated.
NA	11.	If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.
K	12.	ESG-PSA Evaluation Form is completed for the scenario at the applicable facility.

ATTACHMENT 5
ESG CRITICAL TASKS

20-01 NRC Scenario #1

CT-1 (CT-11)

Critical Task

Close containment isolation valves such that at least one valve is closed on each Phase A penetration before transition out of TRIP-1.

Safety Significance

The containment isolation system preserves the ability of the containment boundary to minimize the release of fission products following a LOCA or a fuel handling accident within the containment. The Phase A containment isolation signal serves to actuate the containment isolation system following a LOCA. The Phase A containment isolation signal actuates all power operated valves that are not required to be open for the operation of essential equipment during an accident and that can be immediately closed without increasing the potential for damage to the containment equipment.

Cues:

- Indication and/or annunciation that SI is actuated

AND

- One or more of the following:
  - Absence of annunciation that Phase A isolation is actuated
  - Absence of closed valve position indication on all (both) containment isolation valves on one or more Phase A penetrations
  - Open valve position indication on all (both) containment isolation valves on one or
  - more Phase A penetrations
  - ESF system status lamps show that all (both) containment isolation valves on
  - one or more Phase A penetrations are not closed

Measurable Performance Indicator

Manipulation of controls as required to close at least one containment isolation valve on each Phase A penetration

- Closed valve position indication for at least one containment isolation valve on each Phase A penetration

Feedback

- ESF system status lamps show that at least one containment isolation valve is closed on each Phase A penetration

CT-2 (CT-36)

Critical Task

Transfer to cold leg recirculation before receiving the RWST Lo-Lo alarm or cavitation to ECCS Pumps (this CT includes time critical operator actions per OP-SA-102-106-F1)

Safety Significance

Failure to transfer to cold leg recirculation before the RWST inventory is totally depleted results in the loss of all pumped safety injection and containment spray when the RWST empties.

Cues:

- Indication and/or annunciation that safety injection is actuated

AND

- Indication and/or annunciation that RWST level is at or below the Low Level Alarm

AND

- Indication that containment sump level is at or above the minimum level (62%) required for transfer to cold leg recirculation

Measurable Performance Indicator

Manipulation of controls as required to transfer to cold leg recirculation and establish ECCS recirculation flow that at least meets the assumptions of the plant-specific LOCA analyses:

- Valve position indication that the cold leg recirculation flow path is established
- Control switch indication that the circuit breakers or contactors for the low-head injection pumps (and other ECCS injection pumps as necessary) are closed
- These operator time critical actions are accomplished as follows from the time from RWST Lo Level:
  - Initating close on 2SJ69 ≤ 3.7 minutes
  - Stopping one Containment Spray Pump ≤ 5.5 minutes
  - Switch over to recirculation complete ≤ 11.2 minutes

Feedback

- Flow indication of the recirculation of containment sump water through the RHR heat exchangers and into the RCS

ATTACHMENT 6

ESG – PSA RELATIONSHIP EVALUATION

EVENTS LEADING TO CORE DAMAGE

Y/N	Event	Y/N	Event
N	TRANSIENTS with PCS Unavailable	N	Loss of Service Water
N	Steam Generator Tube Rupture	N	Loss of CCW
N	Loss of Offsite Power	N	Loss of Control Air
N	Loss of Switchgear and Pen Area Ventilation	N	Station Black Out
Y	LOCA		

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
N	Containment Sump Strainers	N	Gas Turbine
N	SSWS Valves to Turbine Generator Area	N	Any Diesel Generator
N	RHR Suction Line valves from Hot Leg	N	Auxiliary Feed Pump
N	CVCS Letdown line Control and Isolation Valves	N	SBO Air Compressor

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
N	Restore AC power during SBO
N	Connect to gas turbine
N	Trip Reactor and RCPs after loss of component cooling system
Y	Re-align RHR system for re-circulation
N	Un-isolate the available CCW Heat Exchanger
N	Isolate the CVCS letdown path and transfer charging suction to RWST
N	Cooldown the RCS and depressurize the system
N	Isolate the affected Steam Generator that has the tube rupture(s)
N	Early depressurize the RCS
N	Initiate feed and bleed

SIMULATOR
EXAMINATION SCENARIO GUIDE

SCENARIO TITLE: NRC-2 [AB.CVC, AR.ZZ-11, AB.SW-1, AB.CN-1, AB.LOAD-1, TRIP-1, LOSC-1, TRIP-3]

SCENARIO NUMBER: 21-01 NRC ESG-2

EFFECTIVE DATE: See Approval Below

EXPECTED DURATION: 75 minutes

REVISION NUMBER: 03

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **STA**

[ ] **OTHER** _____

Revision Summary:

- Rev. 01: Modified for 19-01 ILOT NRC Exam. Incorporated NRC comment on 6-4-20 to swap RC Wide Range Pressure fails low with 21 SW Accumulator pressure low from ESG-1 to prevent overlap concerns with SRO Admin JPM task in ESG-1. Added comments from NRC Prep Week. [Note: not used on 19-01 NRC exam]
- Rev. 02, Modified for 20-01 ILOT NRC Exam.
- Rev. 03 Modifieid for 21-01 ILOT NRC exam

PREPARED BY:	K. Hantho **Lead Regulatory Exam Author**	9-15-22 **Date**
APPROVED BY:	M. Wadusky **Operations Training Manager or designee**	**Date**
APPROVED BY:	M. Winkelspecht **Facility Representative**	**Date**

SCAN OF SIGNED SCENARIO COVER SHEET

Form 3.3-1 Scenario Outline

Facility:	Salem	Scenario #:	2
Scenario Source:	20-01 ESG-2	Op. Test #:	21-01
Examiners:		Applicants/ Operators:	

Initial Conditions: Salem Unit 2 is at 100% power MOL

Turnover: The crew is directed to reduce power to 95% power at 10% per hour IAW S2.OP-IO.ZZ-0004 using boration, control rods and turbine load control in preparation for removing 21 Heater drain pump from service. The 21 Service Water pump is C/T for maintenance.

Critical Tasks:

1. Isolate faulted Steam Generator prior to exiting LOSC-1
2. Terminate safety injection before transition out of TRIP-3

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	ATC(R) BOP(R) CRS(N)	Load reduction to 95% at 10%/hr IAW IOP-4 for heater drain pump removal from service
2	PR0017A	ATC (I,MC) CRS (I,TS)	Pressurizer level control channel fails high
3	N/A	ATC(N) CRS(N)	21 RCP standpipe level low
4	SW0215C SW0339F	BOP (C,MC) CRS (C,TS)	23 SW pump trips and 26 SW pump fails to start.
5	CN0017	ALL(C)	23 Condensate pump trips
6	MS0088Ar	ALL(C)	Steam leak inside containment
7	MS0088Ar	ALL(M)	Steam leak worsens requiring Rx Trip & MSLI (**CT-1 and CT-2**)
8	RP0279A or B	ATC(I,MC) CRS(I)	Train initially selected for MSLI fails to manually actuate other train will actuate MSLI
		ABs	AB.CVC-1 → ARP → AB.SW-1 → AB.CN-1 → AB.LOAD-1 →AB.STM-1
		EOPs	TRIP-1 → LOSC-1 → TRIP-3

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

I. OBJECTIVES

A. Given the order or indications of a charging system malfunction, PERFORM actions as the nuclear control operator to RESPOND to the loss or malfunction of the charging system in accordance with the approved station procedures.

B. Given indication of a loss or malfunction of the Charging system DIRECT corrective action for a Charging System malfunction in accordance with the approved station procedures.

C. Given indication of a loss of a SW Pump, PERFORM actions as the nuclear control operator to RESPOND to the malfunction in accordance with approved station procedures.

D. Given indication of a loss of a SW Pump, DIRECT corrective action for a SW System malfunction in accordance with the approved station procedures.

E. Given indication of a loss of a Condensate Pump, PERFORM actions as the nuclear control operator to RESPOND to the malfunction in accordance with approved station procedures

F. Given indication of a loss or malfunction of the Condensate system DIRECT corrective action for a Condensate System malfunction in accordance with the approved station procedures

G. Given the order or indications of a reactor trip, PERFORM actions as the nuclear control operator to RESPOND to the reactor trip in accordance with the approved station procedures.

H. Given indication of a reactor trip, DIRECT the response to the reactor trip in accordance with the approved station procedures

I. Given the order or indications of a safety injection PERFORM actions as the nuclear control operator to RESPOND to the safety injection in accordance with the approved station procedures.

J. Given indication of a safety injection DIRECT the response to the safety injection in accordance with the approved station procedures.

K. Given the order or indications of a loss of secondary heat sink, perform actions as the nuclear control operator to RESPOND to the loss of heat sink in accordance with the approved station procedures.

L. Given indication of a loss of secondary heat sink, DIRECT the response to the heat sink loss in accordance with the approved station procedures.

M. During performance of emergency operating procedures, monitor the critical safety function status trees in accordance the EOP in effect.

II. MAJOR EVENTS

1. PZR Level channel I fails high (2LT-459)
2. 21 RCP standpipe level Low
3. 23 SW Pump trips and 26 SW Pump fails to auto start
4. 23 Condensate Pump trips
5. 21 Steam Line break (inside containment)
6. MSLI fails on first train actuated
7. Isolate faulted SG
8. Terminate SI

III. SCENARIO SUMMARY

A. The crew will receive the unit at 100% power MOL. 21 SW Pump is C/T for pump repack.

B. The crew will perform a load reduction to 95% in preparation for removing 21 Heater Drain pump from service for planned maintenance.

C. Following the down power the controlling PZR level channel will fail high. The crew will respond per **S2.OP-AB.CVC-0001**, Loss of Charging, and take manual control of charging to raise charging flow. The crew will select an operable control channel. The CRS will evaluate Tech Specs and enter 3.3.1.1 action 6.

D. After Tech Specs have been evaluated 21 RCP standpipe level low will alarm, operators will take actions to makeup to the standpipe IAW **S2.OP-AR.ZZ-0011**, Control Console 2CC1 alarm response. Alarm will clear once actions are taken to align flow and complete the makeup to the standpipe.

E. The 23 SW pump will trip and the standby SW pump in Auto (26 SW Pump) will fail to auto start. The crew will manually start 26 SW pump to restore SW header pressure. The CRS may enter **S2.OP-AB.SW-0001**, Loss of SW Header Pressure, but the only action is to start a SW pump. The CRS will evaluate Tech Specs with 21 SW pump C/T and the trip of 23 SW pump and enter TS 3.7.4 for an inoperable SW loop.

F. After Tech Specs are evaluated, the 23 condensate pump will trip. The CRS will enter **S2.OP-AB.CN-0001**, Main Feedwater/Condensate System Abnormality, and take actions to bypass the condensate polisher and initiate a turbine load reduction to 1080 MWe per Attachment 2. The crew performs load reduction IAW **S2.OP-AB.LOAD-0001**, Rapid Load Reduction.

G. After the turbine load reduction is complete, there will be a steam leak inside containment on 21 steamline upstream of the orifice. Operators will observe reactor power rising and containment pressure and temperatures begin to rise. The crew will diagnose a steam leak inside containment and enter **S2.OP-AB.STM-0001,** Excessive steam flow. Actions will need to be taken to maintain power less than 75% limit in place from tripped condensate pump.

H. Once actions have been discussed about an additional downpower then the larger leakrate will be entered requiring the unit to be tripped and MS isolation to be completed. The crew will trip the reactor and enter **2-EOP-TRIP-1**, Reactor Trip or Safety Injection.The crew will initiate MSLI for faulted SG, the first train of main steam isolation selected will fail to actuate, the crew must actuate the other train.

I. Following the MSLI, the crew will recognize that 21 SG is the faulted SG and isolate AFW flow by closing 21AF21 and 21AF11 [**Critical Task #1, this includes a time critical action to isolate AFW flow to the faulted SG by closing the AF21 and AF11 ≤ 10 minutes)**]. The crew will transition to **2-EOP-LOSC-1,** Loss of Secondary Coolant.

J. Once all actions are completed to isolate the faulted SG the crew will transition to **2-EOP-TRIP-3**, Safety injection termination to terminate safety injection [**Critical Task #2**].

K. In **EOP-TRIP-3,** Safety Injection Termination**,** the crew will terminate SI by stopping ECCS pumps, isolating BIT flow and placing normal letdown in service.

L. The scenario may be terminated when the crew places normal letdown in service or at direction by Lead Examiner.

M. INITIAL CONDITIONS

____ **IC-241**

PREP FOR TRAINING (i.e. computer setpoints, procedures, bezel covers ,tagged equipment)

Initial	Description
	VC1and VC4 C/T
	RCPs (SELF CHECK)
	RTBs (SELF CHECK)
	MS167s (SELF CHECK)
	500 KV SWYD (SELF CHECK)
	SGFP Trip (SELF CHECK)
	23 CV PP (SELF CHECK)
	21 SW Pump C/T
	Complete Attachment 2 “Simulator Ready-for-Training/Examination Checklist.”

Note: Tables with blue headings may be populated by external program, do not change column name without consulting Simulator Support group

EVENT TRIGGERS:

Initial	ET #	Description

	1	EVENT ACTION:	kaj02pnh //21WR62 21 RCP STANDPIPE SUPPLY
		COMMAND:	DMF RC009A
		PURPOSE:	<update as needed>
	2	EVENT ACTION:	ka505ava //TRAIN "A" - LOOP 21 MAINSTEAM IS
		COMMAND:	DOR ka705ava
		PURPOSE:	<update as needed>
	3	EVENT ACTION:	ka705ava //TRAIN "B" - LOOP 21 MAINSTEAM IS
		COMMAND:	DOR ka505ava
		PURPOSE:	<update as needed>
	4	EVENT ACTION:	ka505dva //TRAIN "A" - LOOP 22 MAINSTEAM IS
		COMMAND:	DOR ka705dva
		PURPOSE:	<update as needed>
	5	EVENT ACTION:	ka705dva //TRAIN "B" - LOOP 22 MAINSTEAM IS
		COMMAND:	DOR ka505dva
		PURPOSE:	<update as needed>
	6	EVENT ACTION:	ka505hva //TRAIN "A" - LOOP 23 MAINSTEAM IS
		COMMAND:	DOR ka705hva
		PURPOSE:	<update as needed>
	7	EVENT ACTION:	ka705hva //TRAIN "B" - LOOP 23 MAINSTEAM IS
		COMMAND:	DOR ka505hva
		PURPOSE:	<update as needed>
	8	EVENT ACTION:	ka505lva //TRAIN "A" - LOOP 24 MAINSTEAM IS
		COMMAND:	DOR ka705lva
		PURPOSE:	<update as needed>
	9	EVENT ACTION:	ka705lva //TRAIN "B" - LOOP 24 MAINSTEAM IS
		COMMAND:	DOR ka505lva
		PURPOSE:	<update as needed>
	10	EVENT ACTION:	et_array(2).and.ET_array(3) //Event trigger
		COMMAND:	DMF MS0092E
		PURPOSE:	<update as needed>
	11	EVENT ACTION:	et_array(4).and.et_array(5) //Event trigger
		COMMAND:	DMF MS0092F
		PURPOSE:	<update as needed>
	12	EVENT ACTION:	et_array(6).and.et_array(7) //Event trigger
		COMMAND:	DMF MS0092G
		PURPOSE:	<update as needed>
	13	EVENT ACTION:	et_array(8).and.et_array(9) //Event trigger
		COMMAND:	DMF MS0092H
		PURPOSE:	<update as needed>

MALFUNCTIONS:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
_____ 01	PR0017A PZR LEVEL CH I (LT459) FAILS H/L	N/A	N/A	N/A	RT-1	100
_____ 02	RC009A 21 RC PUMP - #3 SEAL FAILS - w/ DEGRADATION	N/A	N/A	N/A	RT-2	100
_____ 03	SW0339F 26 SW PUMP- Press SW Fails H/L	N/A	N/A	N/A	N/A	PRESS SW FAIL HI
_____ 04	SW0215C 23 SERVICE WATER PUMP TRIP	N/A	N/A	N/A	RT-3	
_____ 05	CN0117C 23 CONDENSATE PUMP TRIP	N/A	N/A	N/A	RT-4	
_____ 06	MS0088Ar 21 Main Steam Line Leak Inside Cont BEFORE orifice	N/A	N/A	N/A	RT-5	1
07	RP0279A AUTO MSLIS FAILS TO ACT, TRN A	N/A	N/A	N/A	N/A	
08	RP0279B AUTO MSLIS FAILS TO ACT, TRN B	N/A	N/A	N/A	N/A	
_____ 09	MS0092E 21MS167 FAILS OPEN	N/A	N/A	N/A	N/A	
_____ 10	MS0092F 22MS167 FAILS OPEN	N/A	N/A	N/A	N/A	
_____ 11	MS0092G 23MS167 FAILS OPEN	N/A	N/A	N/A	N/A	
_____ 12	MS0092H 24MS167 FAILS OPEN	N/A	N/A	N/A	N/A	

REMOTES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition
01	SW32D 23 SW PUMP BKR CONTROL POWER	00:00:15	N/A	N/A	RT-9	OFF
02	SW31D 23 SW PUMP RACK OUT	N/A	N/A	N/A	RT-9	TAGGED
_____ 03	MS05A 21MS45 21 STM GEN STM SUP-23 AFP	N/A	100	00:01:00	RT-11	0

OVERRIDES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity
_____ 01	CG01 F DI KCG01TC1 21MS167 21 SG STOP VALVE CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 02	CG02 F DI KCG02TC1 22MS167 22 SG STOP VALVE CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 03	CG03 F DI KCG03TC1 23MS167 23 SG STOP VALVE CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 04	CG04 F DI KCG04TC1 24MS167 24 SG STOP VALVE CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 05	BD12 B DI KBD12DC1 21MS167 HYDRAULIC CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 06	BE12 B DI KBE12DC1 22MS167 HYDRAULIC CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 07	BM12 B DI KBM12DC1 23MS167 HYDRAULIC CLOSE	N/A	N/A	N/A	N/A	OFF
_____ 08	BN12 B DI KBN12DC1 24MS167 HYDRAULIC CLOSE	N/A	N/A	N/A	N/A	OFF

OTHER CONDITIONS:

	Description

SEQUENCE OF EVENTS

a. State shift job assignments.

b. Hold a shift briefing, detailing instruction to the shift: (provide crew members a copy of the shift turnover sheet).

c. Inform the crew “The simulator is running. You may commence panel walkdowns at this time. SM please inform me when your crew is ready to assume the shift”.

d. Allow sufficient time for panel walk-downs. When informed by the SM that the crew is ready to assume the shift, ensure the simulator is cleared of unauthorized personnel.

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
1. Load Reduction to Remove 21 Heater Drain Pump from Service:			
	Crew briefs reactivivty plan to reduce load to 95%.		
	RO initiates a boration and/or inserts control rods.		
	PO initiates a turbine load reduction to 95%.		
Proceed on to next event following load reduction or as directed by Lead Examiner.			
2. Pressurizer Level CH 1 fails high:			
Simulator Operator: Insert **RT-1** at direction from Lead Examiner. **PR0017A, PZR LEVEL CH I (LT459) fails H/L** **Value = 100**			
	RO reports unexpected alarms OHA E-4 PZR LVL HI, E-20, PZR HTR ON LVL HI.		
	PO refers to 2CC1 ARP		
	RO reports PZR Level Channel I has failed high.		
	RO reports charging flow has lowered.		
	RO places Master Flow Control in Manual and raises charging flow to restore PZR level to		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	program.		
	CRS enters S2.OP-AB.CVC-0001		
	PO initiates Attachment 1 CAS.		
	RO reports charging pump is running and no signs of cavitation.		
	RO reports that a PZR level controlling channel has failed high.		
	CRS directs RO to take manual control of charging (if not already performed)		
	RO selects operable Channel 3 for control and an operable Channel 2 or 3 for recorder.		
	RO reports letdown in service.		
	CRS directs PO to review S2.OP-RPS-0003 to remove the failed channel from service. (**No action by PO**)		
TS evaluation #1:			
	CRS enters: • **TS 3.3.1.1 action 6 (72 hours to place the channel in the tripped condition)**		
Proceed to next event after Tech Specs has been evaluated or by direction from Lead Examiner.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
3. 21 RCP standpipe level low			
Simulator Operator: Insert **RT-2** on direction from Lead Examiner. This will insert the following malfunction: **RC009A 21 RC PUMP - #3 SEAL FAILS - w/ DEGRADATION Value = 100**			
	Crew reports unexpected console alarm, 21 RCP Standpipe level low.		
	Crew refers to alarm response and makes up to standpipe and monitors for leakage.		
Simulator Operator: Verify ET-1 is true when 21 WR62 is opened, this will delete MALF RC009A			
	Crew will monitor parameters and possibly initiate a Reactor coolant leakrate surveillance procedure.		
Proceed to next event when direction by Lead Examiner.			
4. 23 SW Pump trips and 26 SW fails to Auto start:			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
Simulator Operator: Enter **RT-3** on direction from Lead Examiner. **SW0215C, 23 SW pump trips** **SW0339F, 26 SW pressure switch fails high**			
	RO reports unexpected OHA alarms for: • B-13, 21 SW HDR PRESS LO • B-14, 22 SW HDR PRESS LO • B-15, TURB AREA SW HDR PRESS LO		
	RO reports 23 SW Pump tripped and 26 SW Pump in Auto failed to start.		
	RO starts 26 SW pump.		
	RO reports low SW pressure alarms are cleared.		
Examiner's Note: Crew does not have to enter S2.OP-AB.SW-0001 since the only action that the AB requires is to start standby SW Pump.			
	CRS enters S2.OP-AB.SW-0001, Loss of SW Header Pressure		
	PO initiates Attachment 1 CAS		
	PO reports no indication of SW Bay leak.		
	PO starts SW Pumps to maintain header pressure between 95-150 psi.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO reports all SW OHA alarms are cleared.		
	PO reports no indication of leak or valve malfunction on the turbine header.		
Role Play: IF an operator is dispatched to investigate the trip of 23 SW Pump, then report back in 2 minutes the following: "the 4KV breaker for 23 SW Pump is open and no relay flags are up", if dispatched to SW report no abnormalities locally at the pump after 4 minutes. Insert **RT-9** to tag SW pump if requested			
TS evaluation #2:			
	CRS enters: • **TS 3.7.4 (72 hours). This is due to having 21 and 23 SW Pumps inoperable in one bay resulting in only one operable SW loop (refer to S2.OP-SO.SW-0005 Attachment 2)**		
Proceed to next event at Lead Examiner direction.			
5. 23 Condensate Pump trips:			
Simulator Operator: Insert **RT-4** by direction from Lead Examiner. **CN0117C, 23 condensate pump**			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
trips			
	PO reports 23 Condensate Pump tripped.		
	CRS enters S2.OP-AB.CN-0001, Main Feedwater/Condensate Abnormality.		
	PO initiates Attachment 1 CAS		
	PO reports SGFP did not trip		
	PO reports 23 Condensate Pump tripped.		
Role Play: IF an operator is dispatched, then report the following after 1-2 minutes; "***23 Condensate Pump breaker is open and the overcurrent relay flag is up".***			
	PO reports SGFP suction pressure is less than 320 psig.		
	PO opens 21-23 CN108's (Polisher Bypass valves)		
	PO reports 2CN47 is not open.		
	CRS evaluates plant conditions IAW Attachment 2 section 4.0		
	CRS determines that a load reduction to 85% Rx power at ≤ 5%/min is required.		
	CRS brief load reduction and Reactivity plan.		
	RO initiates boration IAW S2.OP-SO.CVC-0006,		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	Boration Concentration Control.		
	PO initiates load reduction using turbine load control IAW S2.OP-SO.TRB-0002, Turbine-Generator Shutdown Operations		
	CRS enters S2.OP-AB.LOAD-0001, Rapid Load Reduction.		
	PO initiates Attachment 1 CAS		
	RO maintains Tavg on program IAW Attachment 3		
	CRS directs WCC to transfer heating steam to Unit 1 IAW SC.OP-SO.HS-0001.		
Proceed to next event after the load reduction is complete or at Lead Examiner's direction.			
6. 21 SG Steam Line Break (inside containment):			
Simulator Operator: Insert **RT-5** by direction from Lead Examiner. **MS0088Ar 21 Main Steam Line Leak Inside Cont BEFORE orifice** **Value = 1**			
	RO reports unexpected alarms for C-38 CFCU LK DET HI, CONT PRESS HI, and then shortly after C-30 for CFCU LK DET HI-HI.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO determines no rise in CNTMT radiation and that leak is inside containment based on rising containment pressure.		
	RO reports C-2 CNTMT SUMP PUMP START		
Evaluator's Note: Around this time Cntmt Pressure is rising quickly and Rx power is slowly rising.			
	CRS enters S2.OP.AB.STM-0001, Excessive Steam Flow.		
	CRS directs initiation of Attachment 1 of S2.OP-AB.STM-0001.		
	PO verifies EHC system operating correctly, and MS10s and Steam Dumps are closed.		
Evaluator's Note: It will be apparent quickly that a unit shutdown will not be performed before a Rx trip is required based on rising containment pressure.			
	PO reports no SG Safety Valve is open or leaking		
	RO reports Rx Power is not stable.		
	RO reports normal cntmt radiation levels AND the following alarms and indications: ▪ Rising Cntmt pressure ▪ OHA C-38 CFCU LK DET HI ▪ OHA C-30 CFCU Lk DET HI-HI		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	▪ OHA C-2 CNTMT SUMP START		
	RO places all CFCUs in SLOW speed		
	RO reports steam leak in containment is not isolable		
	CRS determines a Unit shutdown is required and plans rate and reactivity management strategy.		
	RO calculates boration required.		
Record time when steam leak event is inserted: **Time:_____;_____;_____**			
7. Unisolable steam leak inside containment (Initiate Rx Trip, MSLI and SI):			
	CRS determines need to implement CAS of AB.STM-1 based on CNTMT Pressure rising towards Auto SI (4 psig)		
	RO Trips the Rx and confirms Rx is Tripped		
	RO manually initiates MSLI.		
Simulator Operator: Once 21 MS167 is closed then modify **MALF MS0088Ar to 20%**.			
	RO verifies steam leak is not isolated		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	RO initiates SI.		
	RO reports steam leak is on 21 SG.		
	PO throttles AFW flow to no less than 22E4 lbm/hr.		
Critical Task # 1 (CT-17): Isolate feed and steam flow to faulted SG before transition out of EOP-LOSC-1, Loss of Secondary Coolant. This CT also includes a time critical action for isolating AFW feedflow to the ruptured SG by closing the respective AF21 and AF11 ≤ 10 minutes. **SAT_____ UNSAT_____** This CT is comprised of 2 parts; isolate feed flow (Part 1) and then steam flow (Part 2) later in EOP-LOSC-1.			
	Critical Task #1 Part 1 - PO closes 21AF21 and 21AF11 to isolate faulted 21 SG.		
Record time AFW valves are closed: **Time:_____:_____:_____**			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	RO reports SEC loading is not complete for B vital bus, but that all available equipment started.		
	PO reports 21 and 22 AFW pumps are operating.		
	RO reports 21 and 22 CA330s are closed.		
	RO reports containment pressure has NOT remained less than 15 psig.		
	PO reports main steam line isolation requirements were met and MSLI previously initiated.		
	PO reports all 4KV vital busses are energized.		
	RO reports CAV is in Accident Pressurized mode.		
	RO reports correct switchgear room ventilation operation.		
	RO reports 2 CCW pumps are running.		
	RO reports both CCW HXs are in Auto.		
	RO reports correct ECCS pump alignment and expected flows for RCS conditions.		
Examiner's Note: PO should have lowered AFW flow to as close to 22E4 lbm/hr as possible.			
	PO reports AFW flow is >22E4 lbm/hr.		
	RO reports all RCPs are stopped because of phase B		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	actuation.		
	RO reports Tavg is not stable and is rising.		
	RO reports Tavg is trending to 547 F.		
	RO reports both PZR PORVs are closed and block valves are open.		
	PO reports 21 SG pressure is lowering in an uncontrolled manner or completely depressurized.		
	RO reports that MSLI was previously initiated.		
	CRS transitions to 2-EOP-LOSC-1, Loss of Secondary Coolant.		
	RO reports MSLI has been initiated.		
	PO reports all valves in Table A are closed.		
	PO reports 21 SG pressure is dropping in an uncontrolled manner.		
	PO reports that all SGs pressures are NOT dropping in an uncontrolled manner OR completely depressurized.		
	Critical Task #1 - PO closes or verifies closed the following valves: 21BF13, 21BF40, 21MS7, 21MS10, 21MS18, 21MS167, and 21GB4 [Critical Task #2, Part 2 complete]		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
Critical Task #1, Part 2 (CT-17): Isolate feed and steam flow to faulted SG before transition out of EOP-LOSC-1, Loss of Secondary Coolant. **SAT_____ UNSAT_____**			
	PO reports that 21 SG is faulted.		
	PO reports that both 21 and 23 SGs are NOT faulted.		
	CRS dispatches operator to close 21MS45.		
Simulator Operator: Insert **RT-11** to close 21MS45. **MS05A 21 STM GEN SUP-23 AFP** **Final = 0** **Ramp = 1 min**			
	RO resets Phase A.		
	RO opens 21 and 22 CA330s.		
	RO resets SG sample isolation bypass.		
	RO opens 21 thru 24 SS94's.		
	CRS directs Chemistry to sample 21-24 SGs for activity.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO reports all radiation monitors in Table B are Normal.		
	PO reports no SG sample radiation results are available.		
	RO reports RCS subcooling is > 0 F.		
	PO reports that total AFW flow is > 22E4 lbm/hr		
	RO reports that RCS pressure is stable or rising.		
	RO reports that PZR level is > 11% (19% adverse).		
8. SI Termination EOP-TRIP-3:			
	CRS transitions to EOP-TRIP-3, SI Termination.		
	RO resets SI & SEC, Phase A, and Phase B		
Critical Task-3, Part 1 (CT-48): Terminate ECCS flow before transition out of EOP-TRIP-3. **SAT_____ UNSAT_____** **Shaded steps are required to satisfy CT**			
	RO Stops all but one Charging Pump [Critical Task-2, Part 1]		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
Critical Task-2, Part 1 (CT-48): Secure all but 1 charging Pump before transitrion out of EOP-TRIP-3 **SAT_____ UNSAT_____** **Shaded steps are required to satisfy CT**			
	RO reports RCS pressure is stable or rising.		
	RO reports Charging pump suction is aligned to RWST.		
	RO opens 2CV139 and 2CV140		
Critical Task-2, Part 2 (CT-48): Isolate the BIT by closing 2SJ4 and 2SJ5 or 2SJ12 and 2SJ13 **SAT_____ UNSAT_____** **Shaded steps are required to satisfy CT**			
	RO closes 2SJ4, 2SJ5, 2SJ12 and 2SJ13 [Critical Task-2, Part 2]		
	RO closes 2CV71		
	RO opens 2CV68 and 2CV69		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	RO adjusts 2CV55 to maintain PZR level and RCP Seal Injection		
	RO reports PZR level is stable or rising		
	RO reports RCS pressure is stable or rising		
	RO reports RCS pressure is > 1540 psig (1660 psig adverse)		
	RO stops 21 and 22 SI pumps [Critical Task 2, Part 3]		
Critical Task-2, Part 3 (CT-48): Secure SI Pumps before transitrion out of EOP-TRIP-3 **SAT_____ UNSAT_____** **Shaded steps are required to satisfy CT**			
	RO reports both RHR pump suction aligned to RWST.		
	RO stops 21 and 22 RHR pumps [Critical Task 2, Part 4]		
Critical Task-2, Part 4 (CT-48): Secure RHR Pumps before transitrion out of EOP-TRIP-3 **SAT_____ UNSAT_____** **Shaded steps are required to satisfy CT**			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
Scenario may be terminated when CT-2 is completed or by direction from Lead Examiner.			

N. SCENARIO REFERENCES

A. Alarm Response Procedures (Various)
B. Technical Specifications
C. Emergency Plan (ECG)
D. OP-AA-101-111-1003, Use of Procedures
E. 2-EOP-TRIP-1, Rx Trip or Safety Injection
F. 2-EOP-LOSC-1, Loss of Secondary Coolant
G. 2-EOP-TRIP-3, SI Termination
H. S2.OP-AB.SW-0001, Loss of SW Header Pressure
I. S2.OP-AB.CVC-0001, Loss of Charging
J. S2.OP-SO.CN-0001, Main Feedwater/Condensate System Abnormality
K. S2.OP-AB.LOAD-0001, Rapid Load Reduction
L. S2.OP-IO.ZZ-0004, Power Operation
M. S2.OP-AR.ZZ-0011, Control Console 2CC1

ATTACHMENT 1 (ESG-2)
UNIT TWO PLANT STATUS
TODAY

MODE: 1 POWER: 100% RCS BORON: 802 MWe 1220

SHUTDOWN SAFETY SYSTEM STATUS (5, 6 & DEFUELED):

NA

REACTIVITY PARAMETERS

MOST LIMITING LCO AND DATE/TIME OF EXPIRATION:

EVOLUTIONS/PROCEDURES/SURVEILLANCES IN PROGRESS:

1. Load reduction to 95% to support removal of 21 Heater Drain Pump

ABNORMAL PLANT CONFIGURATIONS:

CONTROL ROOM:

Unit 1 and Hope Creek at 100% power.
No penalty minutes in the last 24 hrs.

PRIMARY:

SECONDARY:

- Polisher in service
- Blowdown 35K per loop to 23 Condenser

RADWASTE:

No discharges in progress

CIRCULATING WATER/SERVICE WATER:

- 21 SW Pump C/T for pump repack.

ATTACHMENT 2

SIMULATOR READY FOR TRAINING CHECKLIST

____ 1. Verify simulator is in “TRAIN” Load
____ 2. Simulator is in RUN
____ 3. Overhead Annunciator Horns ON
____ 4. All required computer terminals in operation
____ 5. Simulator clocks synchronized
____ 6. All tagged equipment properly secured and documented
____ 7. TSAS Status Board up-to-date
____ 8. Shift manning sheet available
____ 9. Procedures in progress open and signed-off to proper step
____ 10. All OHA lamps operating (OHA Test) and burned out lamps replaced
____ 11. All printers have adequate paper AND functional ribbon
12. Required procedures clean
____ 13. Multiple color procedure pens available
____ 14. Required keys available
____ 15. Simulator cleared of unauthorized material/personnel
____ 16. All charts advanced to clean traces and chart recorders are on.
____ 17. Rod step counters correct (channel check) and reset as necessary
____ 18. Exam security set for simulator
____ 19. Ensure a current RCS Leak Rate Worksheet is placed by Aux Alarm Typewriter
____ with Baseline Data filled out
____ 20. Recording Media available (if applicable)
____ 21. Ensure ECG classification is correct
____ 22. Reference verification performed with required documents available
____ 23. Verify phones disconnected from plant after drill.
____ 24. Ensure ECG related paperwork is marked “For Training Use Only”.

ATTACHMENT 3

CRITICAL TASK METHODOLOGY

In reviewing each proposed CT, the examination team assesses the task to ensure, that it is essential to safety. A task is essential to safety if, in the judgment of the examination team, the improper performance or omission of this task by a licensee will result in direct adverse consequences or in significant degradation in the mitigative capability of the plant.
The examination team determines if an automatically actuated plant system would have been required to mitigate the consequences of an individual's incorrect performance. If incorrect performance of a task by an individual necessitates the crew taking compensatory action that would complicate the event mitigation strategy, the task is safety significant.

I. Examples of CTs involving essential safety actions include those for which operation or correct performance prevents...

- degradation of any barrier to fission product release
- degraded emergency core cooling system (ECCS) or emergency power capacity
- a violation of a safety limit
- a violation of the facility license condition
- incorrect reactivity control (such as failure to initiate Emergency Boration or Standby Liquid Control, or manually insert control rods)
- a significant reduction of safety margin beyond that irreparably introduced by the scenario

II. Examples of CTs involving essential safety actions include those for which a crew demonstrates the ability to...

- effectively direct or manipulate engineered safety feature (ESF) controls that would prevent any condition described in the previous paragraph.
- recognize a failure or an incorrect automatic actuation of an ESF system or component.
- take one or more actions that would prevent a challenge to plant safety.
- prevent inappropriate actions that create a challenge to plant safety (such as an unintentional Reactor Protection System (RPS) or ESF actuation.

ATTACHMENT 4

SIMULATOR SCENARIO REVIEW CHECKLIST

SCENARIO IDENTIFIER: 21-01 NRC Scenario 2 REVIEWER: K. Hantho

Initials		Qualitative Attributes
K	1.	The scenario has clearly stated objectives in the scenario.
K	2.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue crew into expected events.
K	3.	The scenario consists mostly of related events.
K	4.	Each event description consists of: • the point in the scenario when it is to be initiated • the malfunction(s) that are entered to initiate the event • the symptoms/cues that will be visible to the crew • the expected operator actions (by shift position) • the event termination point
K	5.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.
K	6.	The events are valid with regard to physics and thermodynamics.
K	7.	Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.
K	8.	The simulator modeling is not altered.
K	9.	All crew competencies can be evaluated.
K	10.	The scenario has been validated.
N/A	11.	If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.
K	12.	ESG-PSA Evaluation Form is completed for the scenario at the applicable facility.

21-01 NRC Scenario 2

Critical Tasks:

CT-1 (CT-17)

Isolate the faulted SG before transition out of LOSC-1 (this CT includes time critical operator action time for isolating AFW flow to fauled SG ≤ 10 minutes per OP-SA-102-106-F1)

Safety Significance

Failure to isolate a faulted SG that can be isolated causes challenges to CSFs beyond those irreparably introduced by the postulated conditions. Also, depending upon the plant conditions, it could constitute a failure by the crew to "demonstrate the ability to recognize a failure or an incorrect automatic actuation of an ESF system or component."
Failure to isolate a faulted SG can result in challenges to the following CSFs:

- Integrity
- Subcriticality
- Containment (if the break is inside containment)

Cues:

- Both of the following:
  - Indication of uncontrolled depressurization of multiple SGs

  AND
  - Valve position or flow rate indication that:
    - Main steamline isolation has not actuated
    - Feedline isolation has not actuated

OR

- Both of the following:
  - Steam pressure and flow rate indications that make it possible to identify a single SG as faulted

  AND
  - Valve position and flow rate indication that AFW continues to be delivered to the faulted SG

Measurable Performance Indicator

Manipulation of controls as required to isolate the faulted SG

- MSIVs indicate closed
- Indication of feedline isolation
  - Feedwater control valves indicate closed (**respective AF21 and AF11 are closed ≤ 10 minutes**)
  - Feedline isolation valves indicate closed
  - Main feed pumps indicate tripped
- Indication that AFW flow to the faulted SG is stopped
  - AFW flow control valves for faulted SG indicate closed

Feedback

- Any depressurization of intact SGs stops
- Steam flow indication from faulted SG decreases to zero
- RCS cooldown stops
- Main feedwater flow rate indication of zero

- AFW flow rate indication to faulted SG of zero

CT-2 (CT-48)

Critical Task

Terminate ECCS flow before transition out of TRIP-3

Safety Significance

Failure to terminate ECCS flow when SI termination criteria are met causes the PRZR to fill and RCS pressure to increase. Additionally, the unnecessary continuation of ECCS flow needlessly aggravates the thermal stress on the reactor vessel. This constitutes an incorrect performance that causes a "significant reduction of safety margin beyond that irreparably introduced by the scenario."

Cues:

- Indication and/or annunciation of an extreme (RED path) or severe (ORANGE path) challenge to the integrity CSF

AND

- Indication and/or annunciation of reactor trip and SI

AND

- Indication and/or annunciation that SI termination criteria are met

Measurable Performance Indicator

Manipulation of controls as necessary to terminate ECCS flow

- Breaker position lamps for all but one charging/SI pump indicate open

AND

- Breaker position lamps for all high-head ECCS pumps indicate open

AND

- Breaker position lamps for all low-head ECCS pumps indicate open

AND

- Valve position lamps for BIT isolation valves indicate closed

Feedback

- Indication of charging/SI ECCS flow rate through BIT decreasing or zero
- Indication of high-head ECCS flow rate decreasing or zero
- Indication of low-head ECCS flow rate decreasing or zero

ATTACHMENT 6

ESG-PSA RELATIONSHIP EVALUATION

EVENTS LEADING TO CORE DAMAGE

Y/N	Event	Y/N	Event
N	TRANSIENTS with PCS Unavailable	N	Loss of Service Water
N	Steam Generator Tube Rupture	N	Loss of CCW
N	Loss of Offsite Power	N	Loss of Control Air
N	Loss of Switchgear and Pen Area Ventilation	N	Station Black Out
N	LOCA		

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
N	Containment Sump Strainers	N	Gas Turbine
N	SSWS Valves to Turbine Generator Area	N	Any Diesel Generator
N	RHR Suction Line valves from Hot Leg	N	Auxiliary Feed Pump
N	CVCS Letdown line Control and Isolation Valves	N	SBO Air Compressor

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
N	Restore AC power during SBO
N	Connect to gas turbine
N	Trip Reactor and RCPs after loss of component cooling system
N	Re-align RHR system for re-circulation
N	Un-isolate the available CCW Heat Exchanger
N	Isolate the CVCS letdown path and transfer charging suction to RWST
N	Cooldown the RCS and depressurize the system
N	Isolate the affected Steam Generator that has the tube rupture(s)
N	Early depressurize the RCS
N	Initiate feed and bleed

SIMULATOR
EXAMINATION SCENARIO GUIDE

SCENARIO TITLE: NRC-3 [AR.ZZ-11, AB.CVC-1, AB.PZR-1, AB.TRB-1, TRIP-1, FRSM-1 using BAT pumps, TRIP-2]

SCENARIO NUMBER: 21-01 NRC ESG-3

EFFECTIVE DATE: See Approval Below

EXPECTED DURATION: 60 minutes

REVISION NUMBER: 0

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **STA**

[ ] **OTHER** _____

Revision Summary:
- Rev. 0

PREPARED BY:	K. Hantho **Lead Regulatory Exam Author**	9-16-22 **Date**
APPROVED BY:	M. Wadusky **Operations Training Manager or designee**	 **Date**
APPROVED BY:	M. Winkelspecht **Facility Representative**	 **Date**

SCAN OF SIGNED SCENARIO COVER SHEET

Form 3.3-1 Scenario Outline

Facility:	Salem	Scenario #:	3
Scenario Source:	New	Op. Test #:	21-01
Examiners:		Applicants/ Operators:	

Initial Conditions: Unit 2 is at 93% power MOL

Turnover: The crew is directed to raise power to 98% power at 10% per hour IAW S2.OP-IO.ZZ-0004 using dilution, control rods and turbine load control. The 23 Charging pump is C/T for maintenance.

Critical Tasks:

1. Insert negative reactivity into the core by inserting control rods and/or performing emergency boration and dispatching operators to locally trip the reactor and/or the turbine
2. Close pressurizer PORV block valve prior to step 7 of EOP-TRIP-2

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	ATC(R) BOP(R) CRS(N)	Raise load to 98% at 10%/hr IAW IOP-4.
2	CC0172A	ATC (C,MC) CRS (C)	21 CCW Pump trips and 23 CCW fails to Auto start
3	CV0208A	ATC(C) CRS(C,TS)	21 Charging Pump trips
4	PR0016A	ATC(I,MC) CRS(I,TS)	PZR Pressure Controlling Channel fails HIGH
5	TU0083A	All(M)	Main Turbine high vibrations
6	RP058 RP0059A RP0059B RP0060A RP0060B	ATC(C) BOP(C) CRS(C)	Rx fails to trip by all means from the control room
7	RD0061	ATC(I,MC) CRS(I)	Rods speed controller fails to 8 spm **(CT-1)**
8	VL0298	ATC(C,MC) CRS(C)	2PR2 PZR PORV fails open following exiting EOP-FRSM-1**(CT-2)**
		ABs	ARP → AB.CVC-1 → AB.PZR-1 → AB.TURB-1
		EOPs	TRIP-1 → FRSM using BAT pumps → TRIP-2

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

I. OBJECTIVES

A. Given the order, the crew will commence a power ascension IAW S2.OP-IO.ZZ-0004.

B. Given the indication of a CCW component or system malfunction, perform actions to RESPOND to the loss of a CCW pump in accordance with S2.OP-AB.CC-0001.

C. Given the order or indications of a charging system malfunction, perform actions as the nuclear control operator to RESPOND to the loss or malfunction of the charging system in accordance with the approved station procedures.

D. Given indication of a loss or malfunction of the Charging system DIRECT corrective action for a Charging System malfunction in accordance with the approved station procedures.

E. Given a failure of PZR pressure malfuction, the crew will take corrective action, IAW S2.OP-AB.PZR-0001.

F. Given the order or indications of a PZR pressure system malfunction, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with the approved station procedures.

G. Given indication of a PZR pressure problem, system malfunction DIRECT the response to the malfunction in accordance with the approved station procedures.

H. Given the order or indications of a main turbine or generator malfunction, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with the approved station procedures.

I. Given indications of a main turbine or generator malfunction, DIRECT the response to the malfunction in accordance with the approved station procedures.

J. Given the order or indications of an anticipated transient without trip (ATWT), complete actions as the nuclear control operator to PERFORM the immediate response to the ATWT in accordance with the approved station procedures.

K. Given indication of an anticipated transient without trip (ATWT), DIRECT the immediate response to the ATWT in accordance with the approved station procedures.

L. Given the order or indications of a reactor trip, perform actions as the nuclear control operator to RESPOND to the reactor trip in accordance with the approved station procedures.

M. Given indication of a reactor trip, DIRECT the response to the reactor trip in accordance with the approved station procedures.

N. During performance of emergency operating procedures, monitor the critical safety function status trees in accordance the EOP in effect.

II. MAJOR EVENTS

1. Power ascension to 100% at 10%/hour.
2. 21CCW pump will trip requiring manual start of 23 CCW pump
3. 21 Charging pump trips
4. Pressurizer pressure channel 1 fails high
5. Main turbine rising vibrations
6. Reactor fails to TRIP by all means in the control room
7. 2PR2 pressurizer PORV fails open

III. SCENARIO SUMMARY

1. The crew will continue the power ascension to raise reactor power to 98% at 10% per hour. The reactivity plan calls for performing a dilution first, before raising turbine load. After the crew commences the power ascension, Event #2 will be entered.
2. 21 CCW pump will trip. The crew will receive multiple console alarms for pump trip and note backflashing stop pushbutton for 21 CCW pump. The crew will take actions as directed in the Alarm Response Procedure to start 23 CCW pump. Once the 23 CCW pump is started, Event #3 can be entered.
3. 21 charging pump trips. The crew will respond IAW S2.OP-AB.CVC-0001, Loss of Charging, and start the only remaining charging pump. The crew will then restore letdown flow. The CRS will evaluate Tech Specs and enter 3.5.2.a action a, 3.1.2.2.b, and 3.1.2.4. After Tech Specs have been evaluated, Event #4 will be entered.
4. The controlling PZR pressure channel will fail high. The crew will respond per S2.OP-AB.PZR-0001, Pressurizer Pressure malfunction, and take manual control of the master pressure controller. The crew will select an operable control channel. The CRS will evaluate Tech Specs and enter 3.3.1.1 action 6, and 3.3.2.1.b action 19, 3.4.5 action b and 3.2.5.b. After Tech Specs have been evaluated, Event #5 and 6 will be entered.
5. Main turbine vibrations will begin to rise and the crew will enter S2.OP-AB.TRB-0001, Main Turbine abnormality. The crew will recognize the degrading conditions and recommend a downpower to remove the turbine from service.
6. Main turbine vibrations will continue to rise requiring the operators to trip the turbine and reactor based on power level greater than 49%. The crew will perform the trip using guidance in the abnormal procedure and enter 2-EOP-TRIP-1, Reactor Trip or safety injection. The Reactor will fail to trip using the trip switches and reactor trip breaker Bezel pushbuttons requiring entry into 2-EOP-FRSM 1, Response to nuclear power generation-ATWS.
7. While in 2-EOP-TRIP-1, Reactor Trip or safety injection actions will be required to insert negative reactivity and rods will not step in automatically. The crew will be required to take control rods to manual and begin inserting negative reactivity using the Rod control manual pushbutton for rod insertion. Once actions have been completed in 2-EOP-FRSM 1, Response to nuclear power generation-ATWS to establish a boration CT#1 then the reactor trip breakers will be open locally by the dispatched operator. Event #8 will be entered once the transition is made back to 2-EOP-TRIP-2, Reactor Trip response.
8. During EOP-TRIP-2 Reactor trip response, the crew will recognize that 2PR2 is open and attempt to close and when it will not close the crew will close the 2PR7 block valve CT#2.

A. INITIAL CONDITIONS

____ **IC-242**

PREP FOR TRAINING (i.e. computer setpoints, procedures, bezel covers ,tagged equipment)

Initial	Description
____	VC1and VC4 C/T
____	RCPs (SELF CHECK)
____	RTBs (SELF CHECK)
____	MS167s (SELF CHECK)
____	500 KV SWYD (SELF CHECK)
____	SGFP Trip (SELF CHECK)
____	21 CV PP (SELF CHECK)
____	**23 CV Pump C/T**
____	**Suggested Protected Equipment:** ▪ **None**
____	Complete Attachment 2 “Simulator Ready-for-Training/Examination Checklist.”

Note: Tables with blue headings may be populated by external program, do not change column name without consulting Simulator Support group

EVENT TRIGGERS:

Initial	ET #	Description	
	2	EVENT ACTION: COMMAND: PURPOSE:	PCRTIME > 5200 <update as needed>
	3	EVENT ACTION: COMMAND: PURPOSE:	KB440TTD //REACTOR TRIP BREAKER "A"-TRIP <update as needed>

MALFUNCTIONS:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
_____ 01	CC0361C 23 COMPONENT COOLING PUMP Fails to Start on Low Pressure	N/A	N/A	N/A	N/A	
_____ 02	CC0172A 21 COMPONENT COOLING PUMP TRIP	N/A	N/A	N/A	RT-1	
_____ 03	CV0208A 21 CHARGING PUMP TRIP	N/A	N/A	N/A	RT-2	
_____ 04	TU0083A MN TURBINE HI VIBRATION - BRG #	N/A	0	00:05:00	RT-4	10
_____ 05	RP0060A FAILURE OF TRAIN "A" RX TRIP BREAKER TO TRIP	N/A	N/A	N/A	N/A	
_____ 06	RP0060B FAILURE OF TRAIN "B" RX TRIP BREAKER TO TRIP	N/A	N/A	N/A	N/A	
_____ 07	RP0058 FAILURE OF AUTOMATIC RX TRIP	N/A	N/A	N/A	N/A	
_____ 08	RP0059A FAILURE OF MANUAL RX TRIP	N/A	N/A	N/A	N/A	
09	RP0059B FAILURE OF MANUAL SI/RX TRIP	N/A	N/A	N/A	N/A	
_____ 10	RD0061 ROD SPEED CONTROL PROGRAM FAILS	N/A	N/A	N/A	ET-3	8
_____ 11	VL0298 2PR2 Fails to Position (0-100%)	N/A	N/A	N/A	RT-5	100
_____ 12	PR0016A PZR PRESS CH I (PT455) FAILS H/L	N/A	0	00:00:10	RT-3	2500

REMOTES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition
_____ 01	PR34D PORV STOP VALVE 2PR6 TAGGED	N/A	N/A	N/A	RT-13	TAGGED
_____ 02	RP07D 21 MG SET MOTOR BKR	N/A	N/A	N/A	RT-10	STOP
_____ 03	RP08D 22 MG SET MOTOR BKR	N/A	N/A	N/A	RT-10	STOP
04	RP09D 21 MG SET GEN OUTPUT BKR	N/A	N/A	N/A	RT-10	TRIPPED
05	RP10D 22 MG SET GEN OUTPUT BKR	N/A	N/A	N/A	RT-10	TRIPPED
_____ 06	RP18D Open MAIN RX TRIP BKR A	N/A	N/A	N/A	RT-9	TRIP OPEN

_____ 07	RP19D Open MAIN RX TRIP BKR B	N/A	N/A	N/A	RT-9	TRIP OPEN

OVERRIDES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity
_____ 01	C310 F DI KC310TN0 2E6D PRESSURE HEATER BUS 480V-OPEN	N/A	N/A	N/A	N/A	OFF
_____ 02	C510 F DI KC510TN0 2G6D PRESSURE HEATER BUS 480V-OPEN	N/A	N/A	N/A	N/A	OFF

OTHER CONDITIONS:

	Description

SEQUENCE OF EVENTS

a. State shift job assignments.

b. Hold a shift briefing, detailing instruction to the shift: (provide crew members a copy of the shift turnover sheet).

c. Inform the crew “The simulator is running. You may commence panel walkdowns at this time. SM please inform me when your crew is ready to assume the shift”.

d. Allow sufficient time for panel walk-downs. When informed by the SM that the crew is ready to assume the shift, ensure the simulator is cleared of unauthorized personnel.

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
1. Power Ascension to 98%			
	CRS briefs crew on power ascension to 98% at 10%/hour IAW S2.OP-IO.ZZ-0004, Power Operation.		
	RO briefs reactivity plan for power ascension.		
	PO briefs turbine load control plan.		
	RO initiates dilution IAW S2.OP-SO.CVC-0006 or uses control rods.		
	PO initiates turbine load control IAW S2.OP-SO.TRB-0001, Turbine Generator Startup Operation.		
	RO monitors Tavg and control rods for proper response.		
Proceed to next event by direction from Lead Examiner.			
2. 21 CCW pump trips, 23 CCW pump fails to Auto start			
Simulator Operator: Insert **RT-1** at the direction from Lead Evaluator. **CC0172A, 21 CCW pump trip**			
	RO reports unexpected console alarms for 21 CC HDR Pressure Lo and RCP Thermal Barrier Discharge Flow Lo.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	RO reports 21 CCW pump tripped and the standby pump failed to Auto start.		
	CRS enters S2.OP-AB.CC-0001.		
Evaluator's Note: The crew can also enter the console ARP procedure and take actions to start the standby CCW pump.			
	RO places 23 CCW in Manual and starts 23 CCW pump.		
	RO reports Discharge Flow Lo is cleared.		
Proceed to next event at Lead Examiner's direction.			
3. 21 Charging pump trip			
Simulator Operator: Insert **RT-2** on direction from Lead Evaluator. **MALF: CV0208A 21 CHARGING PUMP TRIP**			
	RO reports 21 charging pump trip.		
	RO reports letdown is isolated as expected.		
	CRS enters S2.OP-AB.CVC-0001, Loss of Charging.		
	PO initiates AB.CVC-1 CAS.		
	CRS initiates S2.OP-AB.RCP-0001 from CAS item 4.0.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	(No action will be performed in AB other than monitoring RCP parameters.)		
	RO reports No charging pumps are running.		
	RO reports no indication of cavitation prior to 21 charging pump trip.		
	RO reports letdown is isolated.		
	RO reports 21 Charging pump tripped.		
	RO reports 22 Charging pump is available.		
	RO verifies RCP seal inlet temps <225°F and places 22 Charging pump in service.		
	RO ensures open 2CV7		
Role Play: If dispatched by crew then report back after 2-3 minutes that 21 CVpp breaker is tripped and nothing abnormal noted at the pump.			
	RO opens 2CV2 and 2CV277 and places in Auto.		
	RO opens one letdown orifice and maintains letdown pressure < 300 psig by adjusting 2CV18 in manual.		
	RO places 2CV18 in Auto when one orifice valve is in service.		
	RO reports PZR level can be maintained stable or rising, and returns 2CV55 to auto when PZR level is on		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	program.		
TS evaluation #1:			
	CRS enters TS 3.5.2.a act a (72 hours) 3.1.2.4 (72 hours) and 3.1.2.2.b (72 hours).		
Proceed to next event after Tech Specs has been evaluated or by direction from Lead Examiner.			
4. PZR Pressure Channel 1 Fails High			
Simulator Operator: Insert **RT-3** on direction from Lead Evaluator. **MALF: PR0016A, PZR PRESS CH I FAILS H/L** **Severity: 2500** **Ramp: 10 seconds**			
	RO reports unexpected OHA alarms D-8 RC Press HI, and E-42 2PR1 ½ Trip.		
	RO determines actual pressure is not high and reports spray valves open and heaters off, and recommends placing Master Pressure Controller in manual.		
	CRS directs RO to place Master Pressure Controller (MPC) in manual.		
	RO places MPC to Manual and lowers demand (increase pressure) to close sprays and energize PZR heaters.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS enters S2.OP-AB.PZR-0001, Pressurizer Pressure Malfunction.		
	CRS directs initiation of Att. 1 CAS.		
	CRS gives band for control of PZR pressure.		
	RO reports PZR Press Channel I failed high.		
	RO selects Channel III for control.		
	RO matches MPC demand to current pressure and returns Master Pressure Controller to AUTO.		
	RO closes 2PR6 to comply with Tech Specs.		
Evaluator's Note: The requirement to shut 2PR6 and deenergize it is within one hour of the PZR Pressure Channel I instrument failure IAW Tech Specs.			
	CRS directs WCC to remove control power from 2PR6.		
Simulator Operator: Insert **RT-10** 2 minutes after being directed to remove power from 2PR6 **PR34D PORV Stop Valve 2PR6 Condition: Tagged**			
	CRS directs PO to initiate removing failed channel from service and contact I&C for support IAW S2.OP-SO.RPS-0003.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO checks that tripping associated bistables will not result in an ESF or RPS actuation.		
Evaluator's Note: Crew may wait for I&C response before performing any actions in S2.OP-SO.RPS-0003.			
	PO verifies that Channel III is selected for Master Pressure Control.		
	PO selects PZR Pressure recorder to channel other than I.		
TS evaluation #1:			
	CRS enters LCOs 3.3.1.1 Action 6 (72 hours), 3.3.2.1.b Action 19* (72 hours), 3.4.5.b (1 hour), and 3.2.5.b DNB (2 hours).		
Proceed to next event after Tech Specs has been evaluated or by direction from Lead Examiner.			
5. Main turbine high vibrations			
Simulator Operator: Insert **RT-4** when directed by lead examiner.			
Evaluators Note: OHA G-35 TSI trouble will alarm with high vibration from malfunction entered			
	PO will announce unexpected alarm G-35 TSI Trouble and dispatch secondary operator to evaluate		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO will check vibrations on plant computer and note that vibrations have risen and continue to rise		
	CRS will hold brief to discuss limits and requirements to trip the turbine when > 9 Mils on any bearing		
	CRS may direct a downpower to reduce the vibrations		
Vibrations will continue to rise and reach the turbine trip required setpoint of > 9 mils			
	CRS will direct reactor trip and turbine trip when vibrations continue to rise and are rising above the trip criteria.		
6. Reactor fails to trip by all means from the control room			
	RO attempts to trip Rx and it fails to trip		
	Crew announces ATWT		
Simulator operator: Ensure **ET-3** is true when RO attempts to open A Rx trip breaker. This will insert **MALF RD0061 ROD SPEED CONTROL PROGRAM FAILS** and fail the auto rod speed insertion rate to 8 steps per minute.			
	RO attempts to trip using Both Rx trip switches and Rx trip breaker bezels		
	RO recommends entering EOP-FRSM-1		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS enters EOP-FRSM-1, Response to Nuclear Power Generation.		
Evaluator note: If RO allows rods to insert in auto the speed will be failed to 8 steps per minute, should recognize failure and place in manual and insert rods.			
	RO initiates manual rod insertion. [This action also satisfies CT-1]		
	The crew dispatches NEO to locally open the Reactor Trip Breakers and M-G Set breakers		
	PO reports PZR heater bus breakers will not open.		
	PO trips the Main Turbine.		
	PO starts 21 and 22 AFW pumps.		
	PO reports three S/G's greater than 14%		
	PO reports total AFW flow.		
	RO reports immediate actions of EOP-TRIP-1 and FRSM-1 are complete.		
	RO ensures 22 Charging pump is running. (21 CV is C/T).		
	RO reports SI has not actuated.		
	RO starts both boric acid pumps in fast speed.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	RO opens 2CV175 Rapid Boration Valve		
	RO closes 21 and 22CV160s		
	RO reports flow is indicated on Rapid Borate flowmeter.		
	RO places 2CV55 or Master flow controller in manual and verifies flow from charging pump is greater than 87 gpm		
Critical Task-1 (CT-52): Insert negative reactivity into to the core by at least one of the following methods, and dispatch operators to locally trip the reactor and/or the turbine: • De-energize RDMG sets • Insert control rods • Establish Emergency Boration flow to the RCS **SAT_____ UNSAT_____**			
Locally Open Rx Trip breakers:			
Simulator Operator: Insert **RT-9** to open both RTBs breakers when directed by lead evaluator. Timeframe from being dispatched may need to be delayed to ensure the evaluators have time evaluate completion of Critical task-1(CT-52) by the			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
crew **NOTIFY** control room when RTBs are open.			
Simulator Operator: Insert **RT-10**. To open the Rod Drive MG set motor and generator breakers.			
	RO reports PZR pressure is less than 2335 psig.		
	RO reports 2VC5 and 2VC6 are closed.		
	RO reports RTBs and bypass breakers are open.		
Evaluator's Note: The crew will continue in FRSM-1 but should recognize that the RTBs are now open and report PRNI < 5% and IR SUR negative (step 7 CAS) and then GO TO step 16 to exit procedure.			
	PO reports all turbine stop valves are closed.		
	RO reports PRNIs indicate < 5%.		
	RO reports IRNI SUR is negative.		
	CRS GOES TO Step 16.		
	CRS directs boration to continue to obtain Adequate SDM		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS returns to procedure in effect, EOP-TRIP-1.		
5. PZR PORV (2PR2) fails open:			
Simulator Operator: Insert **RT-4** at direction from Lead Evaluator. **MALF: VL0298 2PR2 fails to position (0-100%)** **Final Value: 100**			
	RO reports unexpected OHA for 2PR1/2 NOT CLSD.		
	RO reports PZR pressure is not stable or trending to 2235 psig.		
	RO reports PZR pressure is not > 2235 psig.		
	RO reports PZR pressure lowering, and identifies 2PR2 failed open.		
	RO reports 2PR2 will NOT close in manual.		
Critical Task-2 (CT-10): Close the block valve upstream of the stuck open PZR PORV. **SAT_____ UNSAT_____**			
	RO closes PZR PORV block valve 2PR7. [Critical Task-2]		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
The scenario may be terminated after the crew has Completed CT-2 or by direction from Lead Examiner.			

B. SCENARIO REFERENCES

A. Alarm Response Procedures (Various)
B. Technical Specifications
C. Emergency Plan (ECG)
D. OP-AA-101-111-1003, Use of Procedures
E. S2.OP-IO.ZZ-0004, Power Operation
F. S2.OP-AB.CVC-0001, Loss of Charging
G. S2.OP-AB.CN-0001, Main Feedwater/Condensate System Abnormality
H. S2.OP-AB.RCP-0001, RCP Abnormality
I. S2.OP-AR.ZZ-0011, 2CC1 Alarm Response Procedure
J. S2.OP-AR.ZZ-0013, 2CC2 Alarm Response Procedure
K. 2-EOP-TRIP-1, Reactor Trip or Safety Injection
L. 2-EOP-FRHS-1, Response to Loss of Secondary Heat Sink

ATTACHMENT 1 (NRC-3)
UNIT TWO PLANT STATUS
TODAY

MODE: 1 POWER: 93% RCS BORON: 778 MWe 1130

SHUTDOWN SAFETY SYSTEM STATUS (5, 6 & DEFUELED): N/A

REACTIVITY PARAMETERS

- Rx Plan: To raise Rx power to 98% in preparation for calorimetric at 10%/hour add 1000 gallons of water along with withdrawing control rods to maintain Tavg on program. Reactor Engineering directs the crew to perform a dilution first. Fuel is conditioned no rod withdrawal limitations apply.

MOST LIMITING LCO AND DATE/TIME OF EXPIRATION:

- None

EVOLUTIONS/PROCEDURES/SURVEILLANCES IN PROGRESS:

- Continue power ascension to 98% at 10%/hour IAW S2.OP-IO.ZZ-0004, Power Operation
- Engineering is monitoring BF19s vibrations during startup
- .

ABNORMAL PLANT CONFIGURATIONS:

CONTROL ROOM:

Unit 1 and Hope Creek at 100% power.
No penalty minutes in the last 24 hrs.

PRIMARY:

- 23 CV Pump C/T for lube.

SECONDARY:

- Blowdown 35K per loop to 23 condenser / flashtank
- Polisher in service

RADWASTE:

No discharges in progress

CIRCULATING WATER/SERVICE WATER:

None

ATTACHMENT 2

SIMULATOR READY FOR TRAINING CHECKLIST

____ 1. Verify simulator is in “TRAIN” Load

____ 2. Simulator is in RUN

____ 3. Overhead Annunciator Horns ON

____ 4. All required computer terminals in operation

____ 5. Simulator clocks synchronized

____ 6. All tagged equipment properly secured and documented

____ 7. TSAS Status Board up-to-date

____ 8. Shift manning sheet available

____ 9. Procedures in progress open and signed-off to proper step

____ 10. All OHA lamps operating (OHA Test) and burned out lamps replaced

____ 11. Required chart recorders advanced and ON (proper paper installed)

12. All printers have adequate paper AND functional ribbon

____ 13. Required procedures clean

____ 14. Multiple color procedure pens available

____ 15. Required keys available

____ 16. Simulator cleared of unauthorized material/personnel

____ 17. All charts advanced to clean traces and chart recorders are on.

____ 18. Rod step counters correct (channel check) and reset as necessary

____ 19. Exam security set for simulator

____ 20. Ensure a current RCS Leak Rate Worksheet is placed by Aux Alarm Typewriter
____ with Baseline Data filled out

____ 21. Shift logs available if required

____ 22. Recording Media available (if applicable)

____ 23. Ensure ECG classification is correct

____ 24. Reference verification performed with required documents available

____ 25. Verify phones disconnected from plant after drill.

____ 26. Verify EGC paperwork is marked “Training Use Only” and is current revision.

____ 27. Ensure sufficient copies of ECG paperwork are available.

ATTACHMENT 3

CRITICAL TASK METHODOLOGY

In reviewing each proposed CT, the examination team assesses the task to ensure, that it is essential to safety. A task is essential to safety if, in the judgment of the examination team, the improper performance or omission of this task by a licensee will result in direct adverse consequences or in significant degradation in the mitigative capability of the plant.
The examination team determines if an automatically actuated plant system would have been required to mitigate the consequences of an individual's incorrect performance. If incorrect performance of a task by an individual necessitates the crew taking compensatory action that would complicate the event mitigation strategy, the task is safety significant.

I. Examples of CTs involving essential safety actions include those for which operation or correct performance prevents...

- degradation of any barrier to fission product release
- degraded emergency core cooling system (ECCS) or emergency power capacity
- a violation of a safety limit
- a violation of the facility license condition
- incorrect reactivity control (such as failure to initiate Emergency Boration or Standby Liquid Control, or manually insert control rods)
- a significant reduction of safety margin beyond that irreparably introduced by the scenario

II. Examples of CTs involving essential safety actions include those for which a crew demonstrates the ability to...

- effectively direct or manipulate engineered safety feature (ESF) controls that would prevent any condition described in the previous paragraph.
- recognize a failure or an incorrect automatic actuation of an ESF system or component.
- take one or more actions that would prevent a challenge to plant safety.
- prevent inappropriate actions that create a challenge to plant safety (such as an unintentional Reactor Protection System (RPS) or ESF actuation.

ATTACHMENT 4

SIMULATOR SCENARIO REVIEW CHECKLIST

SCENARIO IDENTIFIER: 21-01 NRC Scenario #3 REVIEWER: K. Hantho

Initials		Qualitative Attributes
KH	1.	The scenario has clearly stated objectives in the scenario.
KH	2.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue crew into expected events.
KH	3.	The scenario consists mostly of related events.
KH	4.	Each event description consists of: • the point in the scenario when it is to be initiated • the malfunction(s) that are entered to initiate the event • the symptoms/cues that will be visible to the crew • the expected operator actions (by shift position) • the event termination point
KH	5.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.
KH	6.	The events are valid with regard to physics and thermodynamics.
KH	7.	Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.
KH	8.	The simulator modeling is not altered.
KH	9.	All crew competencies can be evaluated.
KH	10.	The scenario has been validated.
NA	11.	If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.
KH	12.	ESG-PSA Evaluation Form is completed for the scenario at the applicable facility.

ATTACHMENT 5

ESG CRITICAL TASKS

21-01 NRC Scenario 3

CT-1 (CT-52)

Critical Task
Insert negative reactivity into the core by at least one of the following methods, and dispatch operators to locally trip the reactor and/or the turbine:

- De-energize the control rod drive MG sets
- Insert control rods
- Establish emergency boration flow to the RCS

Safety Significance
Failure to insert negative reactivity, under the postulated plant conditions, results in an unnecessary situation in which the reactor remains critical or returns to a critical condition. Performance of the critical task would make the reactor subcritical and provide sufficient shutdown margin to prevent (or at least minimize the power excursion associated with) any subsequent return to criticality.
Failure to insert negative reactivity constitutes misoperation or incorrect crew performance which fails to prevent "incorrect reactivity control (such as failure to initiate emergency boration… or manually insert control rods)."

Cues:

- Indication of ATWS

AND

- Indication that the reactor is not tripped and that a manual reactor trip is not effective

Measurable Performance Indicator

Manipulation of controls in the control room as required to initiate the insertion of negative reactivity into the core

- For scenarios in which it is possible for the crew to de-energize the rod drive MG sets from the control room:
  - Circuit breaker position indication lamps show the appropriate breakers to be open
  - Individual rod position indication system shows control rods to be on the bottom
- For scenarios in which it is possible for the crew to insert control rods:
  - Group and individual rod position indication systems show control rods moving inward
  - Control rod banks sequentially reach core bottom
- For scenarios in which it is possible for the crew to initiate emergency boration
  - Valve position indication that the emergency boration flow path is established
  - Indication that the appropriate pump(s) are running
  - Flow rate indication of emergency boration

Feedback

- Indication of a negative SUR on the intermediate range of the excore NIS
- Indication of less than 5% power on the power range of the excore NIS

CT-2 (CT-10)

Critical Task

Close the block MOV upstream of the stuck open PRZR PORV by completion of the first step in the ERG network that directs the crew to close the block MOV

Safety Significance

Closing the block MOV under the postulated plant conditions constitutes a task that "is essential to safety," because "its improper performance or omission by an operator will result in direct adverse consequences or significant degradation in the mitigative capability of the plant." In particular, the crew has failed to prevent "degradation of any barrier to fission product release." In this case, the RCS fission product barrier can be restored to full integrity simply by closing the block MOV. Therefore, failure to close the MOV also represents a failure by the crew to "take one or more actions that would prevent a challenge to plant safety."

Degradation of the RCS fission product barrier is not initiated by operator action. However, the crew's failure to close the block MOV results in unnecessary continuation of the degraded condition of the RCS barrier.

Cues:

- Valve position indication and/or annunciation that the PRZR PORV is open, even after attempts to close it manually from the control room

AND

- Indication and/or annunciation of decreasing RCS pressure

AND

- Indication and/or annunciation consistent with the discharge of PRZR fluid to the PRT
  - PRT temperature, level, pressure
  - Tailpipe RTDs and/or acoustic monitors

AND

- Valve position indication that the block MOV upstream of the stuck open PRZR PORV is Open

Measurable Performance Indicator

Manipulation of controls as required to close the block MOV for the stuck open PRZR PORV

- Valve position indication that the block MOV upstream of the stuck open PRZR PORV is Closed

Feedback

Valve position indication that the block MOV upstream of the stuck open PRZR PORV is closed

ATTACHMENT 6

ESG-PSA RELATIONSHIP EVALUATION

EVENTS LEADING TO CORE DAMAGE

Y/N	Event	Y/N	Event
N	TRANSIENTS with PCS Unavailable	N	Loss of Service Water
N	Steam Generator Tube Rupture	N	Loss of CCW
N	Loss of Offsite Power	N	Loss of Control Air
N	Loss of Switchgear and Pen Area Ventilation	N	Station Black Out
N	LOCA		

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
N	Containment Sump Strainers	N	Gas Turbine
N	SSWS Valves to Turbine Generator Area	N	Any Diesel Generator
N	RHR Suction Line valves from Hot Leg	**N**	Auxiliary Feed Pump
N	CVCS Letdown line Control and Isolation Valves	N	SBO Air Compressor

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
N	Restore AC power during SBO
N	Connect to gas turbine
N	Trip Reactor and RCPs after loss of component cooling system
N	Re-align RHR system for re-circulation
N	Un-isolate the available CCW Heat Exchanger
N	Isolate the CVCS letdown path and transfer charging suction to RWST
N	Cooldown the RCS and depressurize the system
N	Isolate the affected Steam Generator that has the tube rupture(s)
N	Early depressurize the RCS
N	Initiate feed and bleed

SIMULATOR
EXAMINATION SCENARIO GUIDE

SCENARIO TITLE: NRC-4 [IOP-3, AR.ZZ-3, AB.CVC-1, AB.SW-3, LOPA-1]

SCENARIO NUMBER: 21-01 NRC ESG-4

EFFECTIVE DATE: See Approval Dates

EXPECTED DURATION: 70 minutes

REVISION NUMBER: 05

PROGRAM:

[ ] **L.O. REQUAL**

[X] **INITIAL LICENSE**

[ ] **STA**

[ ] **OTHER** _____

Revision Summary:

Rev. 04 (last used 19-01 NRC) Used for 20-01 NRC exam. Modified to include malfunctions not used on the previous two NRC exams which include; 23A CW traveling screen high DP, 21 CFCU failure to start, PZR spray valve 2PS3 fails to close during depressurization.
Rev. 05 Modified for use in 21-01 NRC exam.

PREPARED BY:	K. Hantho **Lead Regulatory Exam Author**	9-21-22 **Date**
APPROVED BY:	M. Wadusky **Operations Training Manager**	11-29-22 **Date**
APPROVED BY:	M. Winkelspecht **Facility Representative**	11-29-22 **Date**

SCAN OF SIGNED SCENARIO COVER SHEET

Form 3.3-1 Scenario Outline

Facility:	Salem	Scenario #:	4
Scenario Source:	20-01 ESG-4	Op. Test #:	21-01
Examiners:		Applicants/ Operators:	

Initial Conditions: Salem Unit 2 is at 2% power BOL; 21 SGFP in service

Turnover: The crew is directed to continue power ascension to 10% reactor power IAW S2.OP-IO.ZZ-0003 using control rods, steam dumps, and turbine load control.

Critical Tasks:

1. Establish 220,000 lbm/hr AFW flow before steam generator dryout occurs (WR ≤ 11%) during SBO
2. Depressurize the intact SG(s) such that a maximum of 100°F per hour cooldown rate is established and SG pressure(s) does not go below 310 psig

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	ATC(R) BOP(R) CRS(N)	Continue power ascension to 10% IAW IOP-3 and enter MODE 1.
2	VC0311A	ATC (I) CRS (I,TS)	Containment pressure channel fails high
3	CV0035	ATC(C,MC) CRS(C)	Charging Master flow controller fails low
4	SW0216A	BOP(C) CRS(I,TS)	# 2 SW bay leak
5	EL0134	All(M)	Loss of offsite power
6	EL0144 EL0145 EL0163	ATC(C) BOP(C) CRS(C)	Loss of all AC power**(CT-2)**
7	AF0353C	BOP(I,MC) CRS(I)	23 AFW pump fails to start**(CT-1)**
		ABs	IOP-3 → ARP → AB.CVC-1 → AB.SW-2
		EOPs	LOPA-1

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

I. OBJECTIVES

1. Given the order, perform actions to raise reactor power IAW S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load.
2. Given the order or indications of a Containment Pressure Channel malfunction, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with the approved station procedures.
3. Given the order or indications of a charging system malfunction, perform actions as the nuclear control operator to RESPOND to the malfunction in accordance with the approved station procedures.
4. Given the order or indications of a charging system malfunction, DIRECT the response to the malfunction in accordance with the approved station procedures.
5. Given the unit at power and a leak of a Service Water Bay, take corrective action IAW AB.SW-0003.
6. Given the order or indications of a SW Bay leak, perform actions as the nuclear control operator to RESPOND to the malfunction, IAW approved station procedures.
7. Given the order or indications of a SW Bay leak, DIRECT the response to the malfunction IAW approved station procedures.
8. Given the order or indications of a reactor trip, perform actions as the nuclear control operator to RESPOND to the reactor trip in accordance with the approved station procedures.
9. Given indication of a reactor trip, DIRECT the response to the reactor trip in accordance with the approved station procedures.
10. Given the order or indications of a loss of all AC power, complete actions as the nuclear control operator to PERFORM the immediate response in accordance with the approved station procedures.
11. Given the order with the immediate response to a loss of all AC accident completed and no safety injection actuated or required, perform actions as the nuclear control operator to RECOVER from the loss of AC in accordance with the approved station procedures.
12. Given indication of a loss of all AC power, DIRECT the immediate response in accordance with the approved station procedures.
13. Given the plant with the immediate response completed for a loss of all AC power which did not result in a required safety injection, DIRECT the recovery of the loss of AC in accordance with the approved station procedures.

II. MAJOR EVENTS

1. Power Ascension
2. Containment pressure channel 1 fails high
3. Charging master flow controller failure requiring manual control of charging
4. #2 SW bay leak
5. Loss of Offsite power
6. Loss of all AC Power
7. Failure of 23 AFW pump to auto start

III. SCENARIO SUMMARY

1. The crew will take the watch with the unit stable at 2% reactor power during a plant startup, BOL. 21 SGFP is in service and 22 SGFP is not in service. Steam dumps are in Main Steam Pressure Control, Automatic, set at 1000 psig. The crew will be instructed to raise power to 10% and enter Mode 1.
2. The crew will initiate power ascension to 10%, and enter Mode 1, using Main Steam Dumps and control rods IAW **S2.OP-IO.ZZ-0003**, Hot Standby to Minimum Load and S2.OP-SO.MS-0002, Steam Dump System Operation, Attachments 3 or 4.
3. Containment pressure channel 1 will fail high. Operators will respond IAW S2.OP-AR.ZZ-0003, Overhead annunciator window C alarm response procedure. CRS will evaluate TS and Enter TS 3.3.2.1 Act b 16. After Tech Specs has been evaluated, Event #3 will be entered.
4. After the containment Pressure channel is addressed, The Charging Master flow controller fails LOW. The crew will enter S2.OP-AB.CVC-0001, Loss of Charging, the crew will take actions place the master flow controller in manual and adjust flow. The CRS will establish band for operation of the Charging Master flow controller. The crew will hold a brief after band established and AB actions completed, upon completion of the brief or update event #4 will be entered.
5. After the Master flow controller failure is addressed, the number 2 Service water bay will experience a leak and require isolation IAW S2.OP-AB.SW-0003, Service Water bay leak. The crew will start all bay 4 pumps and shutdown all bay 2 pumps, as well as performing isolation of the bay utilizing MOV's operated from the control room. The CRS will evaluate Tech Specs with the Service Water bay isolated and enter TS 3.7.4. The crew will direct removal of a CCHX from service and removal of control power from all bay 2 pumps. After Tech Specs is evaluated, Event #5 can be entered.
6. A loss of offsite power will occur. The crew will experience an automatic trip of the unit 2 reactor when all offsite power is lost. The crew will enter EOP-TRIP -1, Reactor Trip response, and transition to EOP-LOPA-1, Loss of all AC power, when it is recognized that no vital buses have power. In LOPA-1 the crew will take actions to check plant conditions, restore AC power, maintain conditions for optimal recovery, Determine if ELAP declaration is required, Commence cooldown when AC power is not yet available. **(CT-2)**

7. While in the EOP network the 23 AFW pump will fail to start. The 23 AFW pump should automatically start with the Loss of offsite power. The crew will take the actions to start the 23 AFW pump to maintain a source of inventory control for the Steam generators. **(CT-1)**
8. The scenario may be terminated when cooldown has been commenced and CT 2 evaluated or by direction from the Lead Examiner.

IV. INITIAL CONDITIONS

____ IC-243

PREP FOR TRAINING (i.e. computer setpoints, procedures, bezel covers ,tagged equipment)

Initial	Description
____ 1	VC1and VC4 C/T
____ 2	RCPs (SELF CHECK)
____ 3	RTBs (SELF CHECK)
____ 4	MS167s (SELF CHECK)
____ 5	500 KV SWYD (SELF CHECK)
____ 6	SGFP Trip (SELF CHECK)
____ 7	23 CV PP (SELF CHECK)
____ 8	**21 SGFP is in service**
____ 9	**IOP-3 open and complete up to step 4.3.16, Power Operation. Attachment 4 is marked up.**
___ 10	**Steam Dumps are in MS Pressure Mode and Auto, and S2.OP-SO.MS-0002 is open and marked up to step 4.4.1**
___ 11	**Rod control in manual.**
___ 12	Complete Attachment 2 “Simulator Ready-for-Training/Examination Checklist.”

Note: Tables with blue headings may be populated by external program, do not change column name without consulting Simulator Support group

EVENT TRIGGERS:

Initial	ET #	Description	
	1	EVENT ACTION: COMMAND: PURPOSE:	kb423dmj //CHARGING FLOW DEM-MANUAL (MAST DMF CV0035 <update as needed>
	2	EVENT ACTION: COMMAND: PURPOSE:	MONP254 < 10. //CONT ROD BANK C < 10 (RX TRIP) <update as needed>

MALFUNCTIONS:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
_____ 01	VC0311A Containment Presssure CH I (PT948D) FAILS (NR only)	N/A	N/A	N/A	RT-1	55
_____ 02	CV0035 CHRG MASTER FLO CNTRLR FAILS H/L	N/A	0	00:05:00	RT-2	0
03	SW0216A 21 SW HDR LEAK IN SW STRUCTURE	N/A	500	00:03:00	RT-3	8000
04	EL0134 LOSS OF ALL 500KV OFF-SITE POWE	N/A	N/A	N/A	RT-4	
_____ 05	AF0353C 23 AFP FAILURE TO AUTO START ON ANY (ALL) SIGNALS	N/A	N/A	N/A	N/A	
_____ 06	EL0144 LOSS OF 2A 4160V VITAL BUS	N/A	N/A	N/A	ET-2	
_____ 07	EL0145 LOSS OF 2B 4160V VITAL BUS	N/A	N/A	N/A	ET-2	
_____ 08	EL0163 2C EMERG DIESEL GENERATOR TRIP	00:00:20	N/A	N/A	ET-2	
_____ 09	VL0045 2CV116 Fails to Position (0-100%)	N/A	N/A	N/A	RT-13	0
_____ 10	VL0087 2CC131 Fails to Position (0-100%)	N/A	N/A	N/A	RT-13	0

REMOTES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition
_____ 01	SW23D 21 SW PUMP BKR CONTROL POWER	00:00:30	N/A	N/A	RT-10	OFF
_____ 02	SW27D 22 SW PUMP BKR CONTROL POWER	00:01:00	N/A	N/A	RT-10	OFF
_____ 03	SW32D 23 SW PUMP BKR CONTROL POWER	00:01:30	N/A	N/A	RT-10	OFF
_____ 04	DG01D DEENERGIZE "A" SEC CABINET	N/A	N/A	N/A	RT-9	YES
_____ 05	DG02D DEENERGIZE "B" SEC CABINET	00:00:30	N/A	N/A	RT-9	YES
06	DG03D DEENERGIZE "C" SEC CABINET	00:01:00	N/A	N/A	RT-9	YES
_____ 07	AF20D 21 AFW PUMP BKR CONTROL POWER	N/A	N/A	N/A	RT-12	OFF
_____ 08	AF25D 22 AFW PUMP BKR CONTROL POWER	N/A	N/A	N/A	RT-12	OFF
_____ 09	AF28D 22 AFW PUMP BKR LOCAL TRIP	N/A	N/A	N/A	RT-12	TRIP
_____ 10	AF23D 21 AFW PUMP BKR LOCAL TRIP	N/A	N/A	N/A	RT-12	TRIP
_____ 11	CV28A 21CV98 RCP SEAL INJ MAN ISOL	N/A	N/A	N/A	RT-11	0
_____ 12	CV29A 22CV98 RCP SEAL INJ MAN ISOL	N/A	N/A	N/A	RT-11	0
_____ 13	CV30A 23CV98 RCP SEAL INJ MAN ISOL	N/A	N/A	N/A	RT-11	0

_____ 14	CV31A 24CV98 RCP SEAL INJ MAN ISOL	N/A	N/A	N/A	RT-11	0

OVERRIDES:

SELF-CHECK	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity

OTHER CONDITIONS:

	Description
____ 1.	None

V. SEQUENCE OF EVENTS

1. State shift job assignments.
2. Hold a shift briefing, detailing instruction to the shift: (provide crew members a copy of the shift turnover sheet).
3. Inform the crew “The simulator is running. You may commence panel walkdowns at this time. CRS please inform me when your crew is ready to assume the shift”.
4. Allow sufficient time for panel walk-downs. When informed by the CRS that the crew is ready to assume the shift, ensure the simulator is cleared of unauthorized personnel.

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
1. Power Ascension			
Examiner's Note: The crew will be using S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load, to perform the power ascension. Step 4.3.16 provides guidance on raising reactor power using Steam Dumps IAW S2.OP-SO.MS-0002. The crew can use Attachments 3 or 4 of S2.OP-SO.MS-0002 to operate Steam Dumps. The intent is for the crew to enter Mode 1, ≈ 6% Rx power, THEN; proceed to next event.			
Examiner's Note: IF console alarm **RC LOOPS Tavg – Tref DEVIATION** is in at the time the crew takes the watch, the crew will be provided instructions during turnover that verification of Tavg is 541 F once per 30 minutes until alarm is reset in Control Room Narrative Log is being performed by the extra NCO. This alarm will clear during the power ascension into Mode 1.			
	CRS directs power ascension using Main Steam Dumps in MS Pressure Control and control rods.		
Examiner's Note: The CRS will direct the			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
crew in the order in which to raise Rx power by withdrawing control rods or raising steam dump demand first.			
	PO raises steam dump demand IAW S2.OP-SO.MS-0002, section 5.4 using Attachments 3 or 4.		
	RO withdraws control rods at the specified increments to maintain Tave on program.		
Examiner's Note: Program Tavg at 10% Rx power is about 549 F. During simulator runs, Steam Dump pressure setting was about 982 psig for a Rx Power of 5.4%.			
	RO announces when NIS indicates 5% Reactor Power and RECORDs time of Mode 1 entry in Control Room Narrative Log.		
Proceed on to next event when Reactor Power is 6% or by direction from Lead Examiner.			
2. Containment pressure channel fails high:			
Simulator Operator: Insert **RT-1** on direction from Lead Examiner. **VC0311** **A, Contatinment press channel 1** **Value = 55**			
	RO announces unexpected OHA alarm for CNTMT PRESS HI and HI-HI OHA C-6 and C-14		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS places power ascension on hold.		
	RO reports actual containment pressure is normal and appears the channel is failed.		
	PO reviews S2.OP-AR-ZZ-0003, and determines notification must be written and TS evaluation performed		
TS evaluation #1:			
	CRS enters: • **TS 3.3.2.1 Act b 16**		
Proceed to next event after CRS evaluates Tech Specs or by direction from Lead Examiner.			
3. Charging Master Flow Controller fails low			
Simulator Operator: Insert **RT-2** on direction from Lead Evaluator. **MALF: CV0035 CHRG MASTER FLO CNTRLR FAILS H/L** **Starting Value: 37** **Final Value: 0** **Ramp: 5 minutes**			
	RO reports lowering charging flow and/or low RCP seal injection flow.		
	RO diagnoses Master Flow Controller (MFC) output lowering with PZR level on (or below) program.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS directs RO to place Master Flow Controller in manual and restore charging flow.		
Simulator Operator: Ensure **ET-2** is true when MFC is placed in manual this will delete MALF CV0035			
	RO places MFC in manual, and adjusts demand to required flow.		
	CRS enters S2.OP-AB.CVC-0001, Loss of Charging, based on the reduction of charging flow.		
	CRS directs initiation of Attachment 1 CAS.		
	RO reports 23 charging pump in service with no indication of cavitation.		
	RO reports NO PZR level channel failure.		
	RO reports VCT level channel has NOT failed.		
	RO reports NO indication of charging header leak.		
	RO reports charging MFC failed.		
	RO reports manual control of 23 charging pump has been established, and reports PZR level can be maintained stable or rising.		
	CRS assigns responsibility and band for manual PZR level control.		
Proceed to next event at Lead Evaluators direction.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
2. #2 SW Bay Leak			
Simulator Operator: Insert **RT-3** on direction from Lead Evaluator. **MALF: SW0216A, 21 SW LEAK IN SW STRUCTURE** **Ramp = 3 min** **Final Value = 8000**			
Evaluator's Note: The first OHA will annunciate ~1 minute after the leak is inserted.			
	RO reports unexpected OHA alarm B-29, followed shortly by B-13 and B-14.		
	ARP for OHA B29, 21-23 SW PMP SUMP AREA LVL HI, directs entry into S2.OP-AB.SW-0003 if alarm is concurrent with OHA B-13, 21 SW HDR PRESS LO.		
	RO either starts a SW pump based on lowering SW header pressure, or announces the auto start of 25 SW pump when it occurs, and the clearing of the low SW header pressures alarms.		
Evaluator's Note: OHA B-29 also directs entry into S2.OP-AB.ZZ-002, Flooding.			
	CRS enters S2.OP-AB.SW-0003 SW Bay Leak per ARP direction or enters S2.OP-AB.SW-001, Loss of SW Header Pressure, then transitions to S2.OP-AB.SW-0003, and initiates CAS.		
	RO reports both SW Bays are in service and dispatches NEO to investigate 2 Bay alarms.		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO monitors Main Turbine / SGFP temps on SPDS and Plant Computer per CAS.		
Role Play: 2 minutes after being contacted, report as NEO that ***there is a large leak in the back of #2 SW bay. The water level is approx. 2.5 feet above the pump pedestal and rising.*** (Bay is considered disabled at 2 ft.)			
	PO opens 21 and 22 SW23's.		
	PO closes 21 and 22 SW17's.		
	PO starts all Bay 4 SW pumps.		
	PO reports all Bay 4 SW pumps are running.		
	PO stops all Bay 2 SW pumps.		
	Crew dispatches NEO to open control power bkrs for 21-23 SW pumps.		
Simulator Operator: Call for First Check (IF directed), **THEN** insert **RT-10** to open control power. **SW23D, 21 SW pump control power** **SW27D, 22 SW pump control power** **SW32D, 23 SW pump control power** NOTIFY control room after last remote is active and report control power breakers have been opened for 21, 22 and 23 SW pumps.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO closes 21SW22 and 21SW20.		
Role Play: Once 2 SW Bay is isolated, ***NEO reports that the leak has stopped. IF ask about the water level, state the water level is up to the bottom of the motor housing.***			
	PO reports the leak is isolated.		
	PO reports SW remains available to the EDGs.		
	PO reports SW is being supplied to the Turbine Building through 23SW20.		
	PO monitors Main Turbine parameters on SPDS and the Plant Computer.		
	PO initiates action to remove one CCHX from service IAW CAS item 4.0.		
TS evaluation #2:			
	CRS enters LCO 3.7.4 (72 hours) due to having only one operable SW loop.		
	CRS may enters S2.OP-AB.ZZ-002, Flooding, due to flooding in 2 Bay.		
Proceed to next event after Tech Spec and ECG classification has been identified or at Lead Evaluators direction.			
4. Manual Rx Trip AND Loss of All AC Power (LOPA)			
Simulator Operator: Insert **RT-4** at the			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
direction of the lead examiner this will cause the Loss of Offsite power and automatic Reactor trip.			
	CRS directs RO to trip the Rx and perform immediate actions of **EOP-TRIP-1**, Reactor Trip or Safety Injection.		
	RO turns reactor trip switch to backup the automatic Rx Trip.		
Simulator Operator: Ensure **ET-2** is TRUE following the Rx trip. This Inserts the following malfunctions: **EL0144: Loss of 2A 4KV bus** **EL0145: Loss of 2B 4KV bus** **EL0163: 2C EDG trip**			
	RO performs Immediate Actions for EOP-TRIP-1: ▪ Confirms the reactor tripped ▪ Trips the main turbine		
	PO reports NO Vital Busses are Energized and entry into EOP-LOPA-1 required		
	CRS and RO verify performance of immediate actions.		
	CRS transitions to EOP-LOPA-1, Loss of All AC Power.		
	RO performs immediate actions of LOPA-1: ▪ Trip the Reactor ▪ Trip the Turbine		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS and RO verify performance of immediate actions.		
	RO may close 2CV2 and 2CV277		
	RO reports both PORVs closed		
	RO reports Excess letdown is isolated		
	PO reports that 23 AFW pump failed to Auto start.		
CT-1 (CT-23) **Critical Task** Establish 22E4 lbm/hr AFW flow before steam generator dryout occurs (WR ≤ 11%) **SAT_____ UNSAT_____**			
	PO manually starts 23 AFW pump. CT #1		
	PO throttles AFW flow to no less than 22E4 lbm/hr. CT #1		
	CRS assigns CAS for loss of 23 AFW pump		
Evaluator's Note: 2A & 2B 4KV vital buses are deenergized with a bus differential. 2C EDG tripped on overspeed. Crew should determine that 2A and B EDG need to be stopped.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO reports 2A and B EDG running with breaker open and 2C EDG tripped.		
	PO reports no vital buses energized.		
Role Play (2C EDG): IF dispatched, report that ***2C EDG tripped on overspeed and the linkage looks bent***. **Role Play (Maintenance on 2A and B 4KV Bus)**: When contacted about status ***state that you will go to the field to get a status***.			
	RO reports SI has not been actuated.		
	RO actuates SI.		
	CRS assigns CAS for SI reset and AB.SSP-1 action if required.		
Evaluator's Note: Around this point the crew should be discussing strategy to recover a vital bus. 2C EDG tripped on overspeed and wil take some time to troubleshoot. 2A and 2B vital bus is deenergized on Bus Differential protection and therefore not expected back soon. Crew should determine that success path is to restore 2C EDG with maintenance assistance.			
	After 74 seconds a crew member performs reset of SI.		
	PO reports no vital buses energized		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	CRS assigns CAS to stop EDG's without SW once SEC's deenergized.		
	CRS reads Continuous Caution C6-1 When power is restored to any 4KV vital bus. Recovery actions should continue starting with Step 26.		
	CRS dispatches operator to deenergize all SECs.		
Deenergize SECs:			
Simulator Operator: After being contacted, Insert **RT-9** to deenergize all SECs with a time delay. **REMOTES:** **DG01D** Deenergize A SEC **DG02D** Deenergize B SEC **DG03D** Deenergize C SEC **NOTIFY** CRS when complete.			
	CRS directs performance of S2.OP-AB.LOOP-1		
Role Play for AB.LOOP-1 (Attachment 2 Part A): If called by a crew member Section1.A (Field Operator) - After **10-15 minutes** report back as field operator that Steps 1.A.1 thru 1.A.4 are complete. Section 1.B steps are performed by the NCO in the control room if resources available. CRS may direct WCC to do this, if so then report back after 20 minutes that Section 1.A and 1.B are			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
complete (simulate that panels and doors are open)			
	CRS directs WCC to ensure 21 and 22 AFW pump breakers open and remove control power for 21 and 22 AFW pumps.		
Simulator Operator: Insert **RT-12** to open 21 and 22 AFW pump breakers and remove control power to 21 and 22 AFW pumps. **Remote:** **AF20D and AF23D – 21 AFW** **AF25D and AF28D – 22 AFW**			
	Crew waits until all SECs are deenergized.		
	CRS receives report that all SECs are deenergized.		
Stop running EDGs:			
	PO stops 2A and 2B EDG based on no service water pump running IAW CAS of LOPA-1.		
	RO depresses stop pushbutton for all loads in Table A of LOPA-1.		
	Crew waits until report from WCC that 21 and 22 AFW pump breakers are open and control power removed		
Simulator Operator: Notify crew when **RT-12** has been completed to open 21 and 22 AFW pump breakers and remove control power from 21 and 22 AFW pumps.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO reports status of 4 KV Vital Busses and priority should be placed on restoring 2C EDG with help from maintenance.		
	PO reports that no EDG's are available and no EDG's running		
	CRS dispatch operator for local start when EDG becomes available.		
	CRS assigns CAS to PO for vital bus energization, SW pump start, and turbine header isolation.		
Evaluator's Note: There will be no console indications to verify valve positions during LOPA-1 only field reports.			
	PO reports 23 and 24 SPT are deenergized		
	CRS has SM contact ESO for power availability		
	CRS directs crew to restore power IAW S2.OP-AB.LOOP-0001, Loss of Offsite Power, while continuing with LOPA-1.		
	SM discusses with TSC additional power source availability/strategy.		
	CRS dispatches operator for Seal injection isolation and seal return and CCW thermal barrier isolation.		
Simulator Operator: Insert **RT-11** to close 21-24 CV98. This will simulate closure for 2CV83, 2CV89 and 2CV95 to isolate seal injection filters.			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
Remotes: **CV28A – 21CV98 fails to position** **CV29A – 22CV98 fails to position** **CV30A – 23CV98 fails to position** **CV31A – 24CV98 fails to position** **Final = 0** **Ramp: 1 min** **Delay = 0-3 mins**			
Simulator Operator: Insert **RT-13** to close 2CC131 and 2CV116 valves. **VL0045 2CV116 fails to position** **Final = 0** **Ramp = 1 min** **VL0087 2CC131 fails to position** **Final = 0** **Ramp: 01:00 min**			
	Crew contacts WCC/SM/DSM for status of 2C EDG and 2A and 2B Vital buses.		
Simulator Operator: Notify crew that maintenenace has investigated the 2C EDG trip mechanism and believe they can restore it in 2 hours.			
	PO reports valves in Table B are closed except 21-24 MS167 crew closes 21-24 MS167		
	PO places 21-24 BF19 and 40s in manual and closed		
	RO states 21-24 SS94 closed		

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
	PO states 21-24 GB4s closed		
	PO states NO SG pressures are dropping in an uncontrolled manner or completely depressurized		
	PO gives AFW status to CRS and current SG levels, if less than 9% will feed at max rate to allow cooldown to establish greatert than 9% for cooldown later		
	PO reports secondary rad monitors normal		
	CRS dispatches personnel to perform checkoff sheet 3		
	PO reports AFWST greater than 21%		
	PO reports SG levels greater than 9%		
CT-2 (CT-26) Depressurize the intact SG(s) such that all the following limiting conditions are met: • The cooldown rate in all RCS cold legs is maintained less than 100°F per hour • SG depressurization is not initiated until narrow-range level in at least one intact SG is greater than 9%(15% for adverse containment) • If narrow-range level cannot be maintained greater than 9%(15% for adverse containment) in at least one			

Evaluator/Instructor Activity	Expected Plant/Student Response	SBT LOG	Comment
intact SG (after depressurization is commenced), then SG depressurization is stopped until narrow-range level is restored to greater than 9%(15% for adverse containment) in at least one intact SG **SAT_____ UNSAT_____**			
	PO opens 21-24MS10s and begins cooldown at less than 100 degrees per hour.CT #2		
	POmonitors SGWL and stops if required and maintains achieveable cooldown rate.		
The scenario maybe terminated when CT # 2 has been evaluated or as directed by the Lead Examiner.			

VI. SCENARIO REFERENCES

1. Alarm Response Procedures (Various)
2. Technical Specifications
3. Emergency Plan (ECG)
4. OP-AA-101-111-1003, Use of Procedures
5. S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load
6. S2.OP-AB.CVC-0001, Loss of Charging
7. S2.OP-SO.MS-0002, Steam Dump System Operation
8. S2.OP-AB.SW-0003, Service Water Bay Leak
9. 2-EOP-TRIP-1, Reactor Trip or Safety Injection
10. 2-EOP-LOPA-1, Loss of All AC Power

ATTACHMENT 1 (NRC-4)
UNIT TWO PLANT STATUS
TODAY

MODE: 2 POWER: 2% RCS BORON: 1584 MWe 0

SHUTDOWN SAFETY SYSTEM STATUS (5, 6 & DEFUELED):

N/A

REACTIVITY PARAMETERS

- Control Bank D at 120 steps.
- Reactor Engineering directs use of control rods and steam dumps to raise power to 10%. No Fuel Conditioning Limits are imposed until 50%.

MOST LIMITING LCO AND DATE/TIME OF EXPIRATION:

None

EVOLUTIONS/PROCEDURES/SURVEILLANCES IN PROGRESS:

- S2.OP-IO.ZZ-0003, Hot Standby to Minimum Load complete up to Section 4.3, step 4.3.18 and S2.OP-SO.MS-0002, Steam Dump System Operation is open at (4.4.1)
- IF RC Loop Tavg – Tref Deviation console alarm is in, then extra NCO will log Tavg > 541 F once per 30 minutes.
- Crew to continue Power ascension to 10% using control rods and steam dumps, and enter Mode 1.
- Reactor Engineering is standing by to support power ascension.
- Mode 1 entry is authorized.

ABNORMAL PLANT CONFIGURATIONS:

CONTROL ROOM:

Unit 1 and Hope Creek at 100% power.

PRIMARY:

SECONDARY:

- On main feedwater using 21 SGFP; 22 SGFP is Out of Service

RADWASTE:

No discharges in progress

CIRCULATING WATER/SERVICE WATER:

ATTACHMENT 2

SIMULATOR READY FOR TRAINING CHECKLIST

____ 1. Verify simulator is in “TRAIN” Load

____ 2. Simulator is in RUN

____ 3. Overhead Annunciator Horns ON

____ 4. All required computer terminals in operation

____ 5. Simulator clocks synchronized

____ 6. All tagged equipment properly secured and documented

____ 7. TSAS Status Board up-to-date

____ 8. Shift manning sheet available

____ 9. Procedures in progress open and signed-off to proper step

____ 10. All OHA lamps operating (OHA Test) and burned out lamps replaced

____ 11. Required chart recorders advanced and ON (proper paper installed)

____ 12. All printers have adequate paper AND functional ribbon

____ 13. Required procedures clean

____ 14. Multiple color procedure pens available

____ 15. Required keys available

____ 16. Simulator cleared of unauthorized material/personnel

____ 17. All charts advanced to clean traces and chart recorders are on.

____ 18. Rod step counters correct (channel check) and reset as necessary

____ 19. Exam security set for simulator

____ 20. Ensure a current RCS Leak Rate Worksheet is placed by Aux Alarm Typewriter with Baseline Data filled out

____ 21. Shift logs available if required

____ 22. Recording Media available (if applicable)

____ 23. Ensure ECG classification is correct

____ 24. Reference verification performed with required documents available

____ 25. Verify phones disconnected from plant after drill.

____ 26. Verify EGC paperwork is marked “Training Use Only” and is current revision.

____ 27. Ensure sufficient copies of ECG paperwork are available.

ATTACHMENT 3

CRITICAL TASK METHODOLOGY

In reviewing each proposed CT, the examination team assesses the task to ensure, that it is essential to safety. A task is essential to safety if, in the judgment of the examination team, the improper performance or omission of this task by a licensee will result in direct adverse consequences or in significant degradation in the mitigative capability of the plant.
The examination team determines if an automatically actuated plant system would have been required to mitigate the consequences of an individual's incorrect performance. If incorrect performance of a task by an individual necessitates the crew taking compensatory action that would complicate the event mitigation strategy, the task is safety significant.

I. Examples of CTs involving essential safety actions include those for which operation or correct performance prevents...

- degradation of any barrier to fission product release
- degraded emergency core cooling system (ECCS) or emergency power capacity
- a violation of a safety limit
- a violation of the facility license condition
- incorrect reactivity control (such as failure to initiate Emergency Boration or Standby Liquid Control, or manually insert control rods)
- a significant reduction of safety margin beyond that irreparably introduced by the scenario

II. Examples of CTs involving essential safety actions include those for which a crew demonstrates the ability to...

- effectively direct or manipulate engineered safety feature (ESF) controls that would prevent any condition described in the previous paragraph.
- recognize a failure or an incorrect automatic actuation of an ESF system or component.
- take one or more actions that would prevent a challenge to plant safety.
- prevent inappropriate actions that create a challenge to plant safety (such as an unintentional Reactor Protection System (RPS) or ESF actuation.

ATTACHMENT 4

SIMULATOR SCENARIO REVIEW CHECKLIST

SCENARIO IDENTIFIER: 21-01 NRC Scenario #4 REVIEWER: K. Hantho

Initials		Qualitative Attributes
KH	1.	The scenario has clearly stated objectives in the scenario.
KH	2.	The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue crew into expected events.
KH	3.	The scenario consists mostly of related events.
KH	4.	Each event description consists of: • the point in the scenario when it is to be initiated • the malfunction(s) that are entered to initiate the event • the symptoms/cues that will be visible to the crew • the expected operator actions (by shift position) • the event termination point
KH	5.	No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.
KH	6.	The events are valid with regard to physics and thermodynamics.
KH	7.	Sequencing/timing of events is reasonable, and allows for the examination team to obtain complete evaluation results commensurate with the scenario objectives.
KH	8.	The simulator modeling is not altered.
KH	9.	All crew competencies can be evaluated.
KH	10.	The scenario has been validated.
NA	11.	If the sampling plan indicates that the scenario was used for training during the requalification cycle, evaluate the need to modify or replace the scenario.
KH	12.	ESG-PSA Evaluation Form is completed for the scenario at the applicable facility.

ATTACHMENT 5

ESG CRITICAL TASKS

21-01 NRC Scenario #4

Critical Tasks:

CT-1 (CT-23)

Critical Task
Establish 220,000 lbm/hr AFW flow before steam generator dryout occurs (WR ≤ 11%) during SBO

Safety Significance
Establishing the minimum required AFW flow rate, under the postulated plant conditions, constitutes a task that "is essential to safety," because "its improper performance or omission by an operator will result in direct adverse consequences or significant degradation in the mitigative capability of the plant."

Failure to establish minimum AFW flow (under the postulated conditions) is a violation of the basic objective of ECA-0.0 and of the assumptions of the analyses upon which ECA-0.0 is based. Both intend to mitigate deterioration of RCS conditions while ac emergency power is not available. Without AFW flow, the SGs could not support any significant plant cooldown. Thus, the crew would lose the ability to delay the adverse consequences of core uncovery. Also without AFW flow, decay heat would still open the SG safety valves and would rapidly deplete the SG inventory, leading to a loss of secondary heat sink, or SG dryout. Decay heat would then increase RCS temperature and pressure until the pressurizer PORVs open, imposing a larger LOCA than RCP seal leakage. Both of these examples violate the basic assumptions of the analyses on which ECA-0.0 is based, complicating the mitigation actions.

Cues:
- Indication and/or annunciation of station blackout

AND
- Indication and/or annunciation that insufficient AFW flow to SGs is present

Measurable Performance Indicator

Manipulation of controls in the control room as required to establish the minimum required AFW flow rate to the SGs
- Valve position indication

OR
- Pump speed increasing

Feedback

Indication that at least the minimum required AFW fl ow is being delivered to the SGs

ATTACHMENT 5

ESG CRITICAL TASKS

CT-2 (CT-26)

Critical Task

Depressurize the intact SG(s) such that all the following limiting conditions are met:

- The cooldown rate in all RCS cold legs is maintained less than 100°F per hour
- SG depressurization is not initiated until narrow-range level in at least one intact SG is greater than 9%(15% for adverse containment)
- If narrow-range level cannot be maintained greater than 9%(15% for adverse containment) in at least one intact SG (after depressurization is commenced), then SG depressurization is stopped until narrow-range level is restored to greater than 9%(15% for adverse containment) in at least one intact SG
- SG pressure does not decrease to less than 310 psig
- RCS cold leg temperature does not decrease to less than 280 Degrees F
- If a positive SUR is indicated on either the source range or the intermediate range (after depressurization is commenced), then SG depressurization is stopped and the RCS is allowed to heat up

Safety Significance

Failure to depressurize the secondary and cool down the RCS, under the postulated plant conditions, results in a greater possibility for core damage. Depressurization and cooldown provide the following benefits, which extend the time to core uncovery and maximize the time available to restore ac power:

- Reduction in the rates of RCS inventory loss and RCP seal degradation
- Replenishment (temporary) of RCS inventory by injection of accumulator liquid

Failure to perform the critical task denies these benefits, reducing the time available to restore AC power before the core uncovers.

Failure to depressurize the secondary and cool down the RCS results in “adverse consequence(s)” and in a “significant degradation in the mitigative capability of the plant.” Such a failure constitutes misoperation or incorrect crew performance which leads to “degradation of {the RCS} barrier to fission product release” and to a “significant reduction of safety margin beyond that irreparably introduced by the scenario.”

Cues:

- Indication and/or annunciation that all ac emergency buses are de-energized
  - Bus-energized lamps extinguished
  - Circuit breaker position
  - Bus voltage
  - EDG status

AND

- Step 17 of ECA-0.0 is reached

Measurable Performance Indicator

Manipulation of controls as required to depressurize the intact SG(s) at the highest rate achievable within the 100°F per hour RCS cold leg cooldown limit

ATTACHMENT 5

ESG CRITICAL TASKS

- Valve position indication that the PORV(s) on the intact SG(s) are open the maximum amount consistent with controllability and the 100°F per hour RCS cold leg cooldown limit
- Indication that the RCS cold leg cooldown rate remains less than 100°F per hour Indication that narrow-range level in at least one intact SG is greater than 9%(15% for adverse containment)
- Indication that SG pressure remains above 310 psig
- Indication that RCS cold leg temperature remains above 280 Degrees F
- Indication of a zero or a negative SUR on the source and intermediate ranges

Feedback

- Indication of decreasing pressure in the intact SG(s)
- Indication of decreasing RCS pressure and temperature
- Accumulators inject

ATTACHMENT 6

ESG-PRA RELATIONSHIP EVALUATION

EVENTS LEADING TO CORE DAMAGE

Y/N	Event	Y/N	Event
N	TRANSIENTS with PCS Unavailable	N	Loss of Service Water
Y	Steam Generator Tube Rupture	N	Loss of CCW
N	Loss of Offsite Power	N	Loss of Control Air
N	Loss of Switchgear and Pen Area Ventilation	N	Station Black Out
N	LOCA		

COMPONENT/TRAIN/SYSTEM UNAVAILABILITY THAT INCREASES CORE DAMAGE FREQUENCY

Y/N	COMPONENT, SYSTEM, OR TRAIN	Y/N	COMPONENT, SYSTEM, OR TRAIN
N	Containment Sump Strainers	N	Gas Turbine
N	SSWS Valves to Turbine Generator Area	N	Any Diesel Generator
N	RHR Suction Line valves from Hot Leg	N	Auxiliary Feed Pump
N	CVCS Letdown line Control and Isolation Valves	N	SBO Air Compressor

OPERATOR ACTIONS IMPORTANT IN PREVENTING CORE DAMAGE

Y/N	OPERATOR ACTION
N	Restore AC power during SBO
N	Connect to gas turbine
N	Trip Reactor and RCPs after loss of component cooling system
N	Re-align RHR system for re-circulation
N	Un-isolate the available CCW Heat Exchanger
N	Isolate the CVCS letdown path and transfer charging suction to RWST
Y	Cooldown the RCS and depressurize the system
Y	Isolate the affected Steam Generator that has the tube rupture(s)
N	Early depressurize the RCS
N	Initiate feed and bleed

Complete this evaluation form for each ESG

PSEG Nuclear LLC

Job Performance Measure

PERFORM CONTROL ROOM LOGS (MODES 1-4) IAW S2.OP-DL.ZZ-0003

JPM Number: 21-01 NRC RO-A1.a

Revision Number: 0

Date: 11/9/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	11/9/22
Validated By:	**E. Gallagher / K. Saulsbery** **SME or Instructor**	**Date:**	11/21/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	11/9/22	New JPM.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC	1. Task description and number, JPM description and number are identified.
RC	2. Knowledge and Abilities (K/A) references are included.
RC	3. Performance location specified. (in-plant, control room, or simulator)
	4. Initial setup conditions are identified.
RC	5. Initiating and terminating cues are properly identified.
RC	6. Task standards identified and verified by SME review.
RC	7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)
RC	8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:
	Procedure(s) S2.OP-DL.ZZ-0003 Rev. 126 Date Checked: 10/13/22
	9. Pilot test the JPM:
	a. Verify cues both verbal and visual are free of conflict, and
	b. Ensure performance time is accurate.
N/A	10. If the JPM cannot be performed as written with proper responses, then revise the JPM.
N/A	11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11/9/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. Ensure simulator is in IC-246, this is an Admin JPM.
2. Ensure the following overrides are at values desired for the JPM being performed.

ID #	Description	Delay Time	Ramp	RT/ET	CUR/VAL	FINAL VALUE
01	B223 R AO GB223BTI 21 OT DELTA T SETPOINT	N/A	N/A	N/A	75.1	N/A
02	B223 L AO GB223ATI 21 OP DELTA T SETPOINT	N/A	N/A	N/A	67.2	N/A
03	B225 R AO GB223BTI 22 OT DELTA T SETPOINT	N/A	N/A	N/A	78.0	N/A
04	B225 L AO GB223ATI 22 OP DELTA T SETPOINT	N/A	N/A	N/A	70.2	N/A
05	B227 R AO GB223BTI 23 OT DELTA T SETPOINT	N/A	N/A	N/A	80	N/A
06	B227 L AO GB223ATI 23 OP DELTA T SETPOINT	N/A	N/A	N/A	68.1	N/A
07	B229 R AO GB223BTI 24 OT DELTA T SETPOINT	N/A	N/A	N/A	77.1	N/A
08	B229 L AO GB223ATI 24 OP DELTA T SETPOINT	N/A	N/A	N/A	68.2	N/A
09	B224 L AO GB224ATI 21 REACTOR COOLANT LOOP DELTA T	N/A	N/A	N/A	61.3	N/A
10	B226 L AO GB226ATI 22 REACTOR COOLANT LOOP DELTA T	N/A	N/A	N/A	63.9	N/A
11	B228 L AO GB228ATI 23 REACTOR COOLANT LOOP DELTA T	N/A	N/A	N/A	62.1	N/A
12	B230 L AO GB230ATI 24 REACTOR COOLANT LOOP DELTA T	N/A	N/A	N/A	62.9	N/A
13	B234 L AO GB234ATI 21 LOOP AVG TEMPERATURE	N/A	N/A	N/A	573.5	N/A
14	B235 L AO GB235ATI 22 LOOP AVG TEMPERATURE	N/A	N/A	N/A	575.6	N/A
15	B236 L AO GB236ATI 23 LOOP AVG TEMPERATURE	N/A	N/A	N/A	574.3	N/A
16	B237 L AO GB237ATI 24 LOOP AVG TEMPERATURE	N/A	N/A	N/A	573.8	N/A

SPECIAL INSTRUCTIONS

1. Copies of Salem 2 Cycle 26 REM (Rev. 21)
2. Marked up copies of Control Room Logs S2.OP-DL.ZZ-0003, Reactor Coolant Temperatures, pages 22-24
3. Black and Red pens
4. OP-AA-111-101-1001, Rev. 6, USE AND DEVELOPMENT OF OPERATING LOGS

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- During the CRS review of the Control Room Logs, it was discovered that the Reactor Coolant Temperatures section of the Control Room Logs was not completed

INITIATING CUE

- You are directed to complete the log readings IAW S2.OP-DL.ZZ-0003, Control Room Logs (Modes 1-4).
- Notify the CRS (lead examiner) the results of the log readings.

TASK STANDARD:

The task is satisfactorily met when the applicant completes the control room logs, determines that 23 RC Loop OTDT setpoint channel check is outside the REM OTDT Setpoint band, and records the reading as UNSAT.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

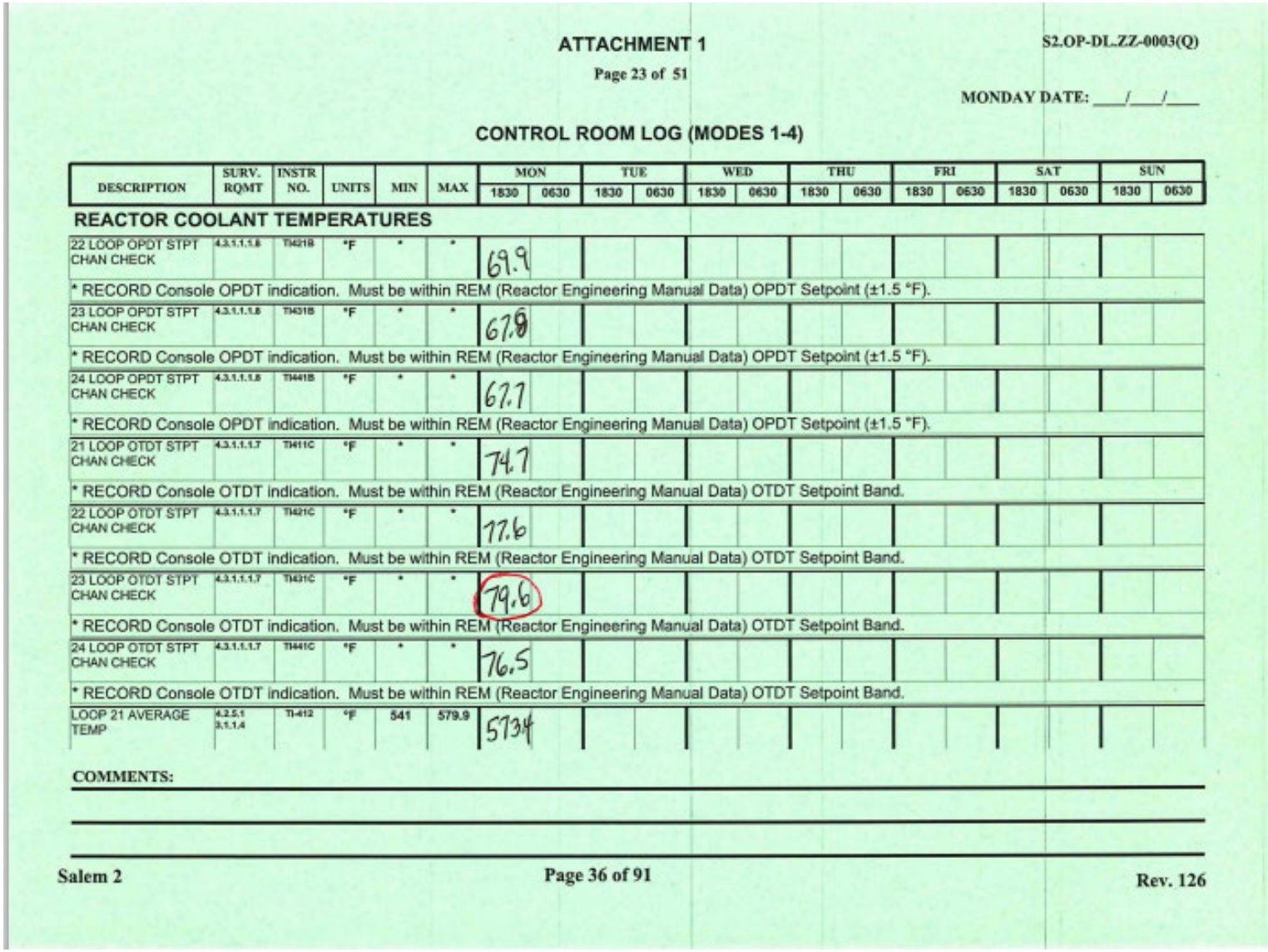
If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):



ATTACHMENT 1 S2.OP-DL.ZZ-0003(Q)

Page 23 of 51

MONDAY DATE: ___/___/___

CONTROL ROOM LOG (MODES 1-4)

DESCRIPTION	SURV. RQMT	INSTR NO.	UNITS	MIN	MAX	MON 1830	MON 0630	TUE 1830	TUE 0630	WED 1830	WED 0630	THU 1830	THU 0630	FRI 1830	FRI 0630	SAT 1830	SAT 0630	SUN 1830	SUN 0630
REACTOR COOLANT TEMPERATURES																			
22 LOOP OPDT STPT CHAN CHECK	4.3.1.1.1.8	TI421B	°F	*	*	69.9													
* RECORD Console OPDT indication. Must be within REM (Reactor Engineering Manual Data) OPDT Setpoint (±1.5 °F).																			
23 LOOP OPDT STPT CHAN CHECK	4.3.1.1.1.8	TI431B	°F	*	*	67.8													
* RECORD Console OPDT indication. Must be within REM (Reactor Engineering Manual Data) OPDT Setpoint (±1.5 °F).																			
24 LOOP OPDT STPT CHAN CHECK	4.3.1.1.1.8	TI441B	°F	*	*	67.7													
* RECORD Console OPDT indication. Must be within REM (Reactor Engineering Manual Data) OPDT Setpoint (±1.5 °F).																			
21 LOOP OTDT STPT CHAN CHECK	4.3.1.1.1.7	TI411C	°F	*	*	74.7													
* RECORD Console OTDT indication. Must be within REM (Reactor Engineering Manual Data) OTDT Setpoint Band.																			
22 LOOP OTDT STPT CHAN CHECK	4.3.1.1.1.7	TI421C	°F	*	*	77.6													
* RECORD Console OTDT indication. Must be within REM (Reactor Engineering Manual Data) OTDT Setpoint Band.																			
23 LOOP OTDT STPT CHAN CHECK	4.3.1.1.1.7	TI431C	°F	*	*	79.6													
* RECORD Console OTDT indication. Must be within REM (Reactor Engineering Manual Data) OTDT Setpoint Band.																			
24 LOOP OTDT STPT CHAN CHECK	4.3.1.1.1.7	TI441C	°F	*	*	76.5													
* RECORD Console OTDT indication. Must be within REM (Reactor Engineering Manual Data) OTDT Setpoint Band.																			
LOOP 21 AVERAGE TEMP	4.2.5.1 3.1.1.4	TI-412	°F	541	579.9	573.4													

COMMENTS:

Salem 2 Page 36 of 91 Rev. 126

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide marked up copy of Control Room Log to applicant. Examiners Note: None Comments: N/A				
1.0	*	Records 21-24 LOOP ACTUAL DT	Determines all LOOP ACTUAL DTs readings are SAT.	
Cue: N/A Examiners Note: None Comments:				
2.0	*	Records LOOP 21-24 OPDT STPT CHAN CHECKs	Determines all OPDT setpoints are within the REM +/- 1.5 °F criteria and Channel Checks are SAT.	
Cue: N/A Examiners Note: None Comments:				
3.0	*	Records LOOP 21-24 OTDT STPT CHAN CHECKs	Determines all OTDT setpoints are within the REM Fig 5A-D and Channel Checks are SAT, **EXCEPT** for 23 OTDT which the setpoint is outside the REM Fig 5C and notes that channel as UNSAT.	
Cue: N/A Examiners Note: UNSAT reading should be red circled Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
4.0	*	Records 21-24 LOOP AVERAGE TEMPs	Determines all LOOP AVERAGE TEMPs are SAT.	
Cue: N/A Examiners Note: None Comments:				
5.0	*	Records LOOP 21-24 AVE TEMP CHANNEL CHECK	Determines 21-24 AVE TEMP CHANNEL CHECK is SAT.	
Cue: N/A Examiners Note: None Comments:				
6.0	*	Records RX VESSEL FLANGE TEMP	Determines RX VESSEL FLANGE TEMP is SAT.	
Cue: N/A Examiners Note: None Comments:				
Terminating Cue		**JPM COMPLETE when the applicant submits the LOGS to the Lead Examiner.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO **X** SRO _____

Facility: Salem **JPM No.:** 21-01 NRC RO-A1.a **Revision No.:** 0

Task Title: Perform Control Room Logs (Modes 1-4) IAW S2.OP-DL.ZZ-0003

Task No.: N1150410501 **Source:** Bank _____ New X Mod _____

System: Conduct of Operations (Generic)

K/A Number / Description: G2.1.25 Ability to interpret reference materials, such as graphs, curves, tables (reference Potential)

K/A Rating RO 3.9 SRO 4.2

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: S2.OP-DL.ZZ-0003 pages 22-24

Reference(s): S2.OP-DL.ZZ-0003 (R126)

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- During the CRS review of the Control Room Logs, it was discovered that the Reactor Coolant Temperatures section of the Control Room Logs was not completed

INITIATING CUE

- You are directed to complete the log readings IAW S2.OP-DL.ZZ-0003, Control Room Log (Modes 1-4).
- Notify the CRS (lead examiner) the results of the log readings.

PSEG Nuclear LLC

Job Performance Measure

DETERMINE TIME TO MAKEUP TO UNIT 2 RWST

JPM Number: 21-01 NRC RO-A1.b

Revision Number: 0

Date: 08/16/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	8/16/22
Validated By:	**J. Militti / J. DeLisle** **SME or Instructor**	**Date:**	8/16/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	8/16/22	New JPM. Converted to new template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-SO.CVC-0006 Rev. 25 Date Checked: 8/16/22
S2.RE-RA.ZZ-0012 Rev. 228
S2.OP-TM.ZZ-0002 Rev. 8

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

N/A 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/16/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. Copies of S2.OP-SO.CVC-0006 (section 5.8.3 only), S2.RE-RA.ZZ-0012, S2.OP-TM.ZZ-0002, and calculator for each applicant.

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- You are the RO on-shift
- Make-up to the RWST is required to restore level to the Tech Spec minimum
- Current RWST level is 40.0 feet
- Current RWST boron concentration is 2360 ppm
- Current BAST boron concentration is 6900 ppm
- Section 5.8.3 of S2.OP-SO.CVC-0006, Boron Concentration Control, will be used to make-up to the RWST.

INITIATING CUE

- The CRS directs you to perform the following:
  1. DETERMINE the boric acid flowrate to make-up to the RWST from the BAST tanks.
  2. DETERMINE the amount of time required to raise the RWST level to the MINIMUM Tech Spec level.
- Record you're response on the Cue Sheet.

TASK STANDARD:

The task is satisfactorily met when the applicant has determined a boric acid flowrate of 26 gpm and will take 59.2 minutes to raise the RWST level to the Tech Spec minimum level.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

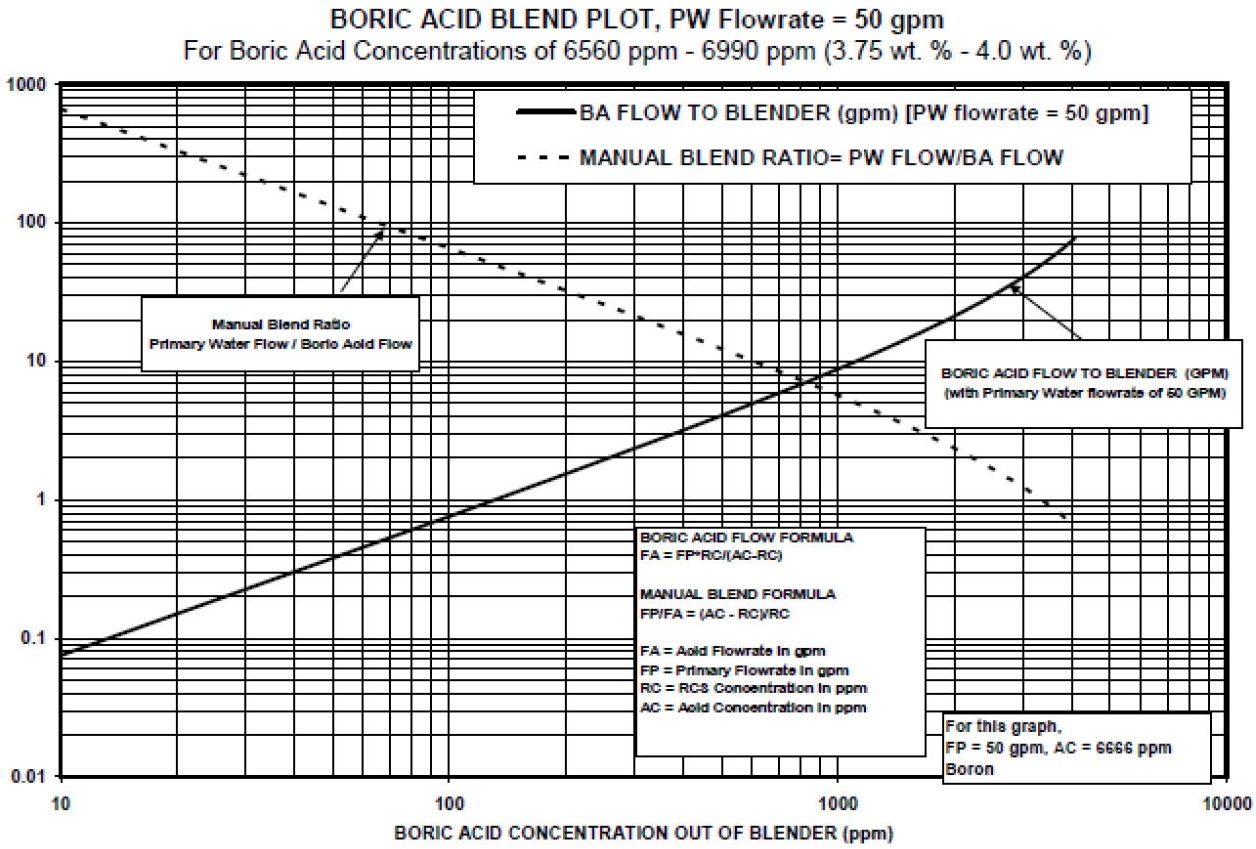
The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

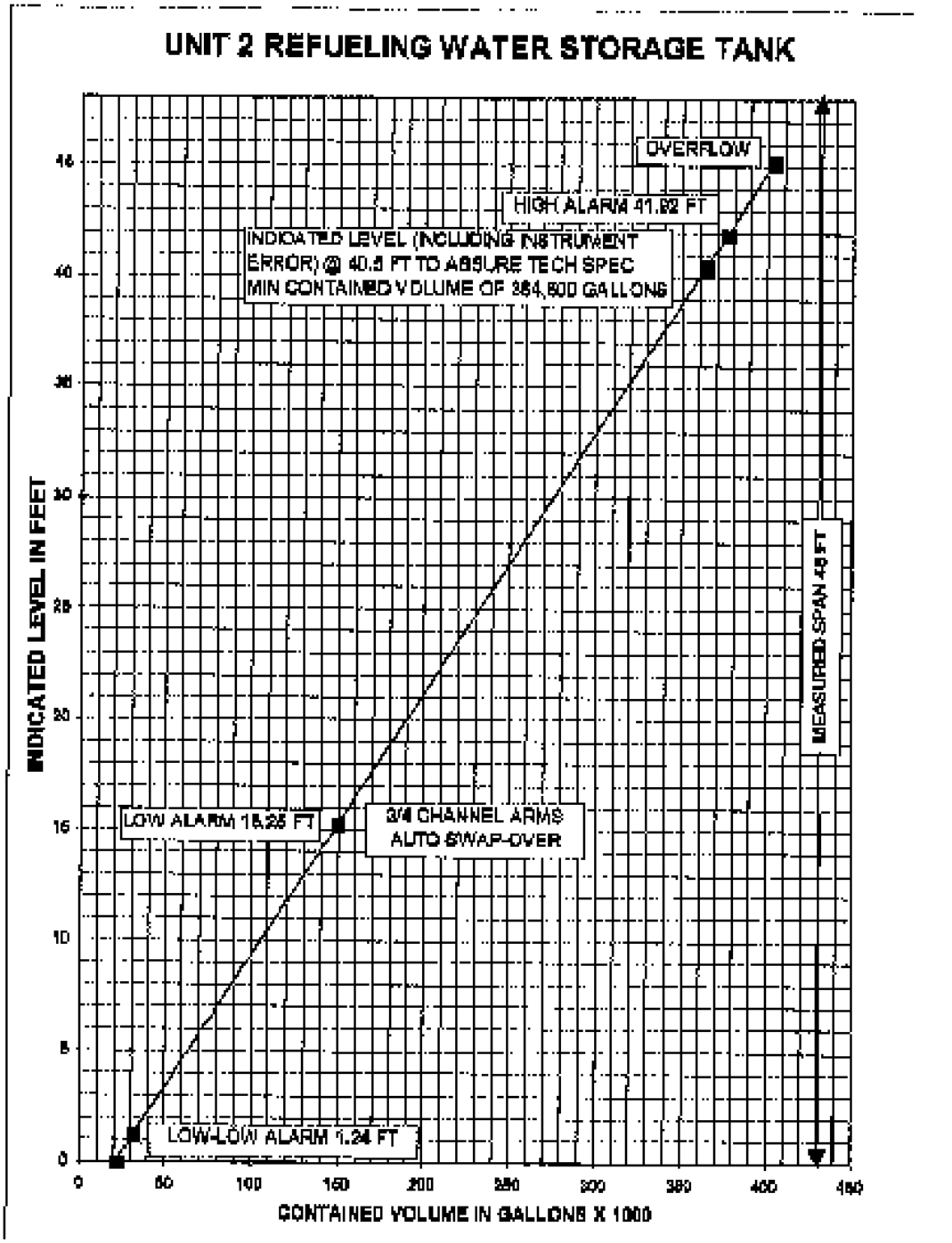
ANSWER KEY (optional):

Attachment 15, Figure 100B - Boric Acid Blend Plot, PW Flowrate = 50 gpm



ANSWER KEY (optional):

S1.OP-TM.ZZ-0002(Q)



Salem Unit 2 Page 28 of 34 Rev. 8

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate copy of S2.OP-SO.CVC-0006, S2.RE-RA.ZZ-0012 (Figures), S2.OP-TM.ZZ-0002 (tank capacity curves), and Calculator. Examiners Note: None Comments: N/A				
1.0	*	Determine boric acid flowrate.	Per step 5.8.3.A, obtain boric flow setpoint from S2.RE-RA.ZZ-0012, Figures 100B or 100D. Determines a boric acid flowrate of 26 gpm using Figure 100B for PW Flowrate = 50 gpm. (50 gpm X 2360 ppm) ÷ (6900 ppm – 2360 ppm) = 26 gpm	
Cue: **WHEN** the applicant is determining boric acid flowrate using the Figure, state the following: Use the formula on the Figures and NOT the curve. **Examiners Note**: Figures 100B and 100D both state that Primary Water (PW) Flowrate is 50 gpm. Auto Make-up typically has the PW Flowrate set to 61-62 gpm (possible error trap). Step 5.8.3.N will direct adjusting the 2CV179 to 50 gpm. Comments:				
2.0	*	Determine the total makeup flowrate.	Determines that the total make-up flowrate is 76 gpm. Boric acid flowrate + primary water flowrate = 26 gpm + 50 gpm = 76 gpm	
Cue: None Examiners Note: None Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
3.0	*	Determines the amount of make-up (in gallons) is required to raise RWTS level from 40.0 feet to Tech Spec minimum level (40.5 feet).	Using Tank Curves for Unit 2 RWST, determines the following: 40.0 feet = 360,000 gallons 40.5 feet = 364,500 gallons 364,500 – 360,000 = 4,500 gallons	
Cue: None Examiners Note: None <u>Comments:</u>				
4.0	*	Determine the amount of time to raise RWST level to the Tech Spec minimum level.	Determines it will take 59.2 minutes to raise the RWST to the minimum Tech Spec level (40.5 feet) 4,500 gallons ÷ 76 gpm = 59.2 minutes	
Cue: None Examiners Note: None <u>Comments:</u>				
Terminating Cue		**JPM COMPLETE when the applicant submits the worksheet.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO **X** SRO _____

Facility: Salem **JPM No.:** 21-01 NRC RO-A1.b **Revision No.:** 0

Task Title: Determine Time to Makeup to Unit 2 RWST

Task No.: N0040170101

Source: Bank _____ New X Mod _____

System: Conduct of Operations (Generic)

K/A Number / Description: G2.1.23 Ability to perform general and/or normal operating procedures during any pant condition

K/A Rating RO 4.3 SRO 4.4

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **15** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: S2.OP-SO.CVC-0006, S2.RE-RA.ZZ-0012, S2.OP-TM.ZZ-0002, Calculator

Reference(s): S2.OP-SO.CVC-0006 S2.RE-RA.ZZ-0012 S2.OP-TM.ZZ-0002

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT/WORKSHEET

Operator's Name: _____

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- You are the RO on-shift
- Make-up to the RWST is required to restore level to the Tech Spec minimum
- Current RWST level is 40.0 feet
- Current RWST boron concentration is 2360 ppm
- Current BAST boron concentration is 6900 ppm
- Section 5.8.3 of S2.OP-SO.CVC-0006, Boron Concentration Control, will be used to make-up to the RWST.

INITIATING CUE

- The CRS directs you to perform the following:
  1. DETERMINE the boric acid flowrate to make-up to the RWST from the BAST tanks.

  2. DETERMINE the amount of time required to raise the RWST level to the MINIMUM Tech Spec level.

RECORD YOU'RE RESPONSE ON THIS WORKSHEET

PSEG Nuclear LLC

Job Performance Measure

PERFORM SEVEN DAY AFD SURVEILLANCE

JPM Number: 21-01 NRC RO-A2

Revision Number: 01

Date: 08/16/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	8/16/22
Validated By:	**J. Militti / J. DeLisle** **SME or Instructor**	**Date:**	8/16/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	8/1/22	Converted to new template. Revised JPM based on Salem 2 Cycle 26 REM data measured 4/25/22.
01	8/16/22	Changed task to perform seven day AFD surveillance based on validator feedback that using Attachment 1 would be better and more operationally valid.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-ST.NIS-0001 Rev. 14
Salem 2 Cycle 26 REM Rev. 17
Salem 2 Cycle 26 COLR Rev. 10

Date Checked: 8/16/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

N/A 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/16/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. MARKED UP copy of S2.OP-ST.NIS-0001, Salem 2 Cycle 26 REM (data measured 4/25/22), Salem 2 Cycle 26 COLR, and a calculator for each applicant.

INITIAL CONDITIONS

- At Time 0830 today, Salem Unit 2 is at 73% power to comply with TS LCO 3.1.3.1 Action c.2 due to a single control rod being declared INOPERABLE

INITIATING CUE

- You are the RO on-shift
- The CRS directs you to perform the Seven Day AFD surveillance test per Section 4.1 of S2.OP-ST.NIS-0001, Power Distribution – Axial Flux Difference
- Current AFD Channel Console Readings are:

  Channel I: -9.5

  Channel II: -10.1

  Channel III: -10.6

  Channel IV: -10.2
- All prerequisites are completed SAT.
- Submit Attachment 1 when complete and print your name.

TASK STANDARD:

The task is satisfactorily met when the applicant has determined that the Seven Day AFD Surveillance is UNSAT because AFD Channel 3 reading is outside the AFD Lower Limit.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

Attachment 1, AFD SEVEN DAY SURVEILLANCE SHEET

DATE:	TIME:

CALCULATION		VALUE
RECORD AFD Target from Reactor Engineering Manual Data Tables, Table 3:		(A) -2.0
RECORD present Reactor Power:		(B) 73 %
CALCULATE AFD Target Value:	[(A) X (B)] ÷ 100% =	(C) -1.5
NOTE The Target Band Upper and Lower Limits shall be the more restrictive of the limits defined in the Core Operating Limits Report (COLR) OR Reactor Engineering Manual Data Tables.		
DETERMINE AFD Upper Limit:	(+) 6 + (C) =	+4.5
DETERMINE AFD Lower Limit:	(−) 9 + (C) =	-10.5

RECORD AFD Channel Readings for each in-service and OPERABLE channel:			
Channel I (N41)	Channel II (N42)	Channel III (N43)	Channel IV (N44)
-9.5	-10.1	-10.6	-10.2

ACCEPTANCE CRITERIA		SAT	UNSAT
At least 3 of 4 NI Channels in-service and OPERABLE.	($)	X	
Each OPERABLE NI Channel reading is between the Upper and Lower Limits, inclusive.	($)		X
NOTE Core Operating Limits Report, Figure 2 is applicable between 50% and 90% Rated Thermal Power. If power is not within this band then mark N/A.			
Each OPERABLE NI Channel is within the Acceptable Operation region of the Core Operating Limits Report, Figure 2.	($)	X	

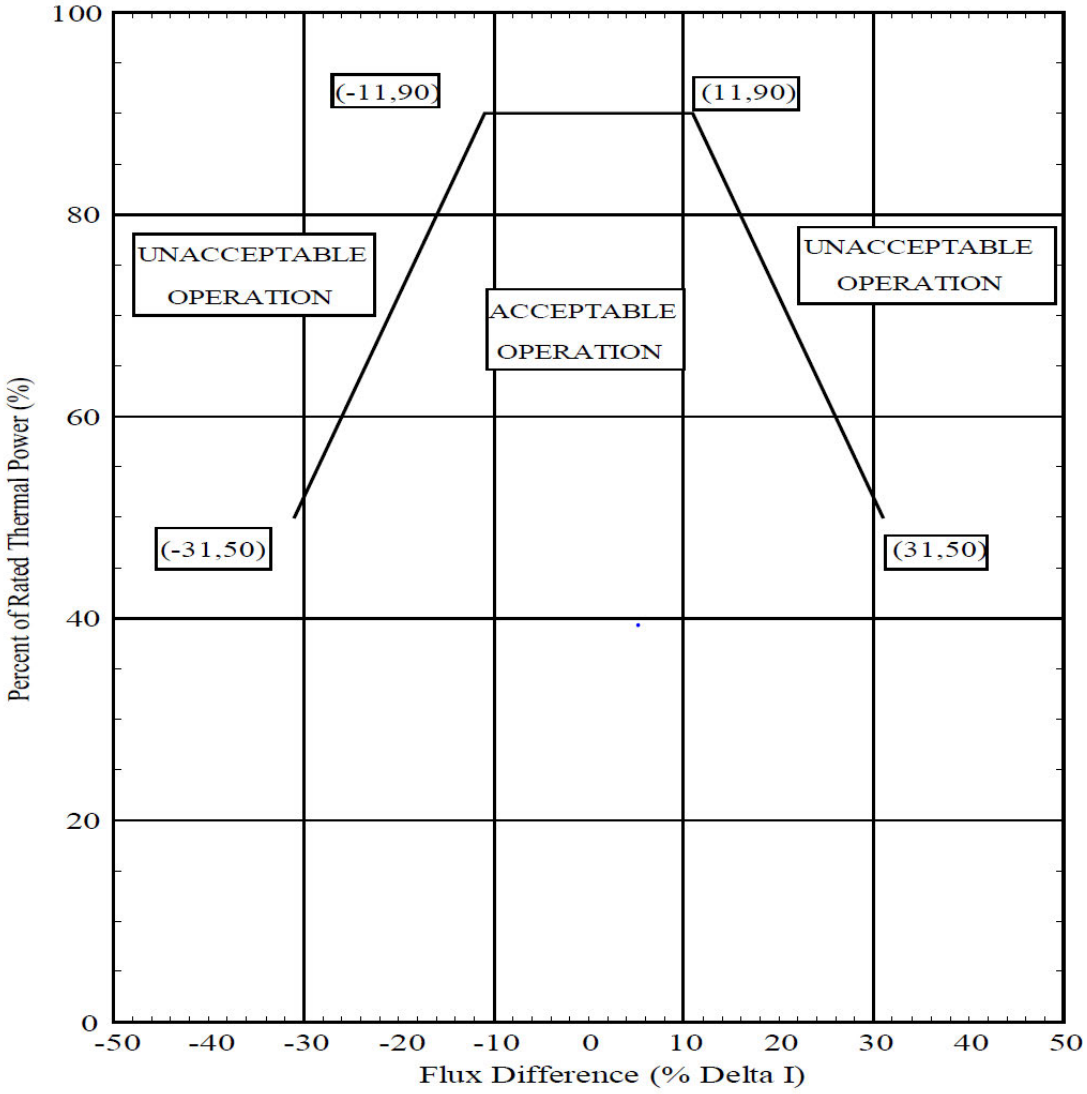
Performed by:	Date:
Verified by:	Date:

Page 1 of 1

ANSWER KEY (optional):

FIGURE 2

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER



ANSWER KEY (optional):

Reactor Engineering Manual Data Salem 2 Cycle 26 Rev 17

TABLE 3
AXIAL FLUX DIFFERENCE TARGET VALUES

Date Measured	Axial Flux Difference Target				Target Band	
	N41	143.7	N43	N44	Upper Limit (+6)	Lower Limit (-9)
BOC	-1.0	-1.0	-1.0	-1.0	5.0	-10.0
11/18/21	-1.5	-1.5	-1.5	-1.5	4.5	-10.5
12/20/21	-1.0	-1.0	-1.0	-1.0	5.0	-10.0
02/22/22	-1.5	-1.5	-1.5	-1.5	4.5	-10.5
04/25/22	-2.0	-2.0	-2.0	-2.0	4.0	-11.0

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide MARKED UP copy of S2.OP-ST.NIS-0001, Salem 2 Cycle 26 REM, Salem 2 Cycle 26 COLR, and Calculator. Examiners Note: None. Comments: N/A				
1	*	RECORD AFD Target from Reactor Engineering Manual Data Tables, Table 3 (A)	Applicant uses Table 3 from the REM and records AFD Target of **-2.0 = (A)**	
Cue: None **Examiners Note**: AFD Target (100% Value) is obtained from Salem 2 Cycle 26 REM Table 3 (data measured 4/25/22) Comments:				
2	*	RECORD present Reactor Power (B)	Applicant records present Reactor Power of **73 = (B)**	
Cue: None Examiners Note: None Comments:				
3	*	CALCULATE the AFD Target Value: [(A) X (B)] ÷ 100% = (C)	Calculates AFD Target Value of **-1.5 = (C)** (C) = -2.0 X 73% ÷ 100% = -1.5	
Cue: None **Examiners Note**: Per P&L 2.1.2, When performing calculations, round value to the first decimal place AND ensure the calculator is cleared between calculations. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		NOTE The Target Band Upper and Lower Limit shall be the more restrictive of the limits defined in the Core Operating Limits Report (COLR) OR Re3actor Engineering Manual Data Tables.		N/A
4	*	DETERMINE AFD Upper Limit: (+) ____ + (C) = Upper Limit	Applicant determines the AFD Target Upper and Lower Limits as: +6.0 + (-1.5) = **+4.5 Upper Limit**	
5	*	DETERMINE AFD Lower Limit: (-) ____ + (C) = Lower Limit	Applicant determines the AFD Target Upper and Lower Limits as: -9.0 + (-1.5) = **-10.5 Lower Limit**	
6		RECORD AFD Channel Readings for each in service and OPERABLE channel: Channel I (N41) = Channel II (N42) = Channel III (N43) = Channel IV (N44) =	Applicant records AFD Channel Readings from the Initiating Cue Channel I (N41) = -9.5 Channel II (N42) = -10.1 Channel III (N43) = -10.6 Channel IV (N44) = -10.2	
Cue: None Examiners Note: None Comments:				
7	*	At least 3 of 4 NI Channels in service and OPERABLE.	Applicant determines all four NI channels are OPERABLE and RECORDs SAT	
Cue: None **Examiners Note**: Cue Sheet does not indicate any NI channels INOPERABLE. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
8	*	Each OPERABLE NI Channel reading is between the Upper and lower Limits inclusive.	Applicant determines that NI Channel 3 is outside the AFD lower limit and RECORDs **UNSAT.**	
Cue: None Examiners Note: None Comments:				
		NOTE Core Operating Limit Report, Figure 2 is applicable between 50% and 90% Rated Thermal Power. If power is not within this band then mark N/A.		N/A
9	*	Each OPERABLE NI Channel is within the Acceptable Operation region of the Core Operating Limits Report, Figure 2.	Applicant determines that each NI Channel is within the Acceptable Operation region of the COLR and RECORDs SAT.	
Cue: None Examiners Note: None Comments:				
Terminating Cue		**JPM COMPLETE when the applicant notifies Lead Examiner of surveillance results.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO **X** SRO _____

Facility: Salem **JPM No.:** 21-01 NRC RO-A2 **Revision No.:** 01

Task Title: Perform Seven Day AFD Surveillance

Task No.: N0150010201

Source: Bank X New _____ Mod _____

System: Equipment Control (Generic)

K/A Number / Description: G2.2.12 Knowledge of surveillance procedures

K/A Rating RO 3.7 SRO 4.1

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: S2.OP-ST.NIS-0001, Unit 2 COLR, Unit 2 REM Data Cycle 26, and Calculator

Reference(s): S2.OP-ST.NIS-0001 Salem Unit 2 COLR Unit 2 REM Data Cycle 26

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT/WORKSHEET

INITIAL CONDITIONS

- At Time 0830 today, Salem Unit 2 is at 73% power to comply with TS LCO 3.1.3.1 Action c.2 due to a single control rod being declared INOPERABLE

INITIATING CUE

- You are the RO on-shift
- The CRS directs you to perform the Seven Day AFD surveillance test per Section 4.1 of S2.OP-ST.NIS-0001, Power Distribution – Axial Flux Difference
- Current AFD Channel Console Readings are:

  Channel I: -9.5

  Channel II: -10.1

  Channel III: -10.6

  Channel IV: -10.2
- All prerequisites are completed SAT.
- Submit Attachment 1 when complete and print your name.

PSEG Nuclear LLC

Job Performance Measure

Activate ERDS IAW EP-SA-325-F8

JPM Number: 21-01 NRC RO-A4

Revision Number: 3

Date: 10/14/2022

Developed By:	**K. Hantho** **Instructor**	**Date:**	10/14/22
Validated By:	**T. Headman (see revision comment)** **SME or Instructor**	**Date:**	11/28/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
03	11/28/22	Used bank JPM. Converted to new JPM template and updated procedure references. Plant computer modifications are in progress in the simulator. EP performed validation of the procedure that activates ERDS and procedure has no issues, however, the ERDS system was not connecting to the NRC. EP is working the issue and expect resolution prior to the next EP drill in December 2022.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) EP-SA-325-F8 R2 Date Checked: 10/14/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11/28/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. See below for special instructions for notifying the NRC

SPECIAL INSTRUCTIONS

1. Copies of EP-SA-325 for each applicant.
2. **PRIOR** to conducting this JPM; **CONTACT** the **NRC ERDS Test monitor** and **NOTIFY** ERDS being used for Training. (Day time: 301-287-9386, Night time: [redacted], Day time cell: [redacted], Erds.resource@nrc.gov)
3. **OPEN** the ERDS program using the SPDS computer terminal in the back computer room. Single click the ERDS program using the mouse, this will now enable the computer in the simulator to connect ERDS to NRC using EP-SA-325-F8.
4. Log onto the ERDS computer in simulator.
5. **ENSURE** the "ERDS XXX" computer is available and working.

When all operators have completed this JPM, THEN:

Simulator Operator:

- **NOTIFY NRC ERDS Test monitor** of that Training evolution has ended.
- **CLOSE** the ERDS program on the SPDS computer in back computer room.

INITIAL CONDITIONS

- Salem Unit 2 experienced a reactor trip and safety injection due to a LOCA
- An **ALERT** has been declared

INITIATING CUE

- You are the Secondary Communicator.
- ACTIVATE ERDS IAW Step 1.2.b of EP-SA-325-F8, Secondary Communicator Log.
- Notify the Shift Manager (Lead Examiner) when ERDS is activated.

TASK STANDARD:

The task is satisfactorily met when the applicant successfully activates ERDS.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

[Page left intentionally blank]

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**Simulator Operator:** • **NOTIFY NRC ERDS Test monitor** of pending Training evolution. • **OPEN** ERDS program on SPDS computer. **LOG ON** to ERDS computer**.**		N/A
Cue: Examiners Note: Comments:				
1.2.b 1.2.b.1	*	Within 60 minutes, **ACTIVATE** ERDS from a **LAN computer:** PROCEED to a computer with its monitor labeled "ERDS XXX" (XXX is the computer's asset tag number) in the Salem Control Room and START the program by SELECTING the ERDS Salem desktop icon.	Operator proceeds to ERDS XXX computer located inside Salem Simulator horseshoe area. Operator selects the ERDS_ Salem desktop icon.	
Cue: Examiners Note: Comments:				
1.2.b.2	*	On the "Login For ERDS Display" popup window **ENTER** the PASSWORD (Password is the same as the Username) and **SELECT** "Login" to proceed.	**Operator enters the PASSWORD and selects "Login" to proceed.**	
Cue: N/A Examiners Note: N/A Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
1.2.b.3	*	SELECT the Action button labeled <CONNECT> on the appropriate Salem 1 or Salem 2 "ERDS Link Control and Status Display Screen".	Operator selects the button labeled <CONNECT> for Salem 2 "ERDS Link Control and Status Display Screen".	
Cue: N/A Examiners Note: N/A Comments:				
1.2.b.4	*	SELECT the <YES> button in the "Current Link Control State is Disconnect. Do You Want To Connect?" box.	Operator selects <YES> button in the "Current Link Control State is Disconnect. Do You Want To Connect?" box.	
Cue: N/A Examiners Note: N/A Comments:				
1.2.b.5	*	OBSERVE that Status changes to "Transmitting Data."	Operator checks that the Status changes to "Transmitting Data"	
Cue: N/A Examiners Note: N/A Comments:				
1.2.b.6	*	VERIFY successful communications status by observing that the "Messages Sent" value is increasing from zero "0" to some positive integer > "0".	Operator verifies that the "Message Sent" value is increasing from zero "0" to some positive integer > "0".	
Cue: N/A Examiners Note: N/A Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
1.2.b.7		INFORM the CRS of successful ERDS activation status OR IF ERDS activation is not successful, THEN CONTACT IT Support: Network Operations (Ext. #7008), EOF IT Supervisor (NETS # 5009), or the IT Help Desk (Ext. 5555) for assistance.	Notifies CRS that ERDS is activated.	
Cue: N/A Examiners Note: N/A Comments:				
Terminating Cue		**JPM COMPLETE when the applicant completes the task.**	**Terminate JPM when operator notifies you that ERDS is activated.**	N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO **X** SRO _____

Facility: Salem **JPM No.:** 21-01 NRC RO-A4 **Revision No.:** 3

Task Title: Activate ERDS per EP-SA-325-F8

Task No.: N1240110501

Source: Bank X New _____ Mod _____

System: Emergency Procedures/Plan

K/A Number / Description: G2.4.43 Knowledge of emergency communications systems and techniques

K/A Rating RO 3.2 SRO 3.8

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator x In-Plant _____ RCA _____

Required Materials: EP-SA-325-F8

Reference(s): EP-SA-325-F8

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT/WORKSHEET

Operator's Name: _____

INITIAL CONDITIONS

- Salem Unit 2 experienced a reactor trip and safety injection due to a LOCA
- An **ALERT** has been declared

INITIATING CUE

- You are the Secondary Communicator.
- ACTIVATE ERDS IAW Step 1.2.b of EP-SA-325-F8, Secondary Communicator Log.
- Notify the Shift Manager (Lead Examiner) when ERDS is activated.

ATTACHMENT 8
SECONDARY COMMUNICATOR LOG

INITIALS

1.2 **For an Alert or higher Emergency Classification Level:**

a. IF an on-shift SRO is not available to fill the OSC Operations Supervisor position, **CALLOUT** an additional SRO and have him/her report to the OSC.

Name: _____ Time: _____ N/A CM2

NOTE

)S Activation and Termination is normally performed by the CM2 in the Control m but if needed, could be performed by any communicator in the TSC or EOF by wing the same directions below and substituting TSC or EOF for Control Room CM2 CM2

b. Within 60 minutes, **ACTIVATE** ERDS from a **LAN computer:**

1. **PROCEED** to a computer with its monitor labeled "**ERDS XXX**" (XXX is the computer's asset tag number) in the Salem Control Room and **START** the program by **SELECTING** the **ERDS_Salem** desktop icon. CM2
2. On the "Login For ERDS Display" popup window **ENTER** the PASSWORD (Password is the same as the Username) and **SELECT** "Login" to proceed. CM2
3. **SELECT** the Action button labeled **<CONNECT>** on the appropriate **Salem 1 or Salem 2** "ERDS Link Control and Status Display Screen". CM2
4. **SELECT** the **<YES>** button in the "Current Link Control State is Disconnect. Do You Want To Connect?" box. CM2
5. **OBSERVE** that Status changes to "Transmitting Data." CM2
6. **VERIFY** successful communications status by observing that the "Messages Sent" value is increasing from zero "0" to some positive integer > "0". CM2
7. **INFORM** the CRS of successful ERDS activation status CM2

OR

IF ERDS activation is not successful, THEN **CONTACT** IT Support: Network Operations (Ext. #7008), EOF IT Supervisor (NETS # 5009), or the IT Help Desk (Ext. 5555) for assistance. CM2

PSEG Nuclear LLC

Job Performance Measure

DETERMINE MINIMUM ECCS INJECTION FLOW

JPM Number: 21-01 NRC SRO-A1.a

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**T. Shetrone /** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	Modified JPM conditions in stem from NRC 17-01 JPM that results in none of the ECCS pumps can be stopped. Converted to new template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

RC Procedure(s) 2-EOP-LOCA-5 Sheet 2 Rev. 41 Date Checked: 8/18/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

N/A 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/18/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. PROVIDE applicants **D size** sheet or greater of 2-EOP-LOCA-5 sheet 2.

INITIAL CONDITIONS

- At Time 0500, Salem Unit 2 was manually tripped and Safety Injection automatically actuated as a result of a LOCA
- At Time 0600, the RWST Low Level Alarm actuates and the crew transitions to 2-EOP-LOCA-3.
- The crew is unable to transfer to Cold Leg Recirculation and transitions to 2-EOP-LOCA-5, Loss of Emergency Recirculation, due to NO recirculation capability.
- ECCS has been reduce to one train with the following ECCS Pump flow rates:
  - 21 SI Pump – 160 gpm
  - 22 Charging Pump – 220 gpm
  - BOTH RHR Pumps were stopped

INITIATING CUE

You are the CRS at Step 16.1 of 2-EOP-LOCA-5.

- DETERMINE the following:
  1. At Time 0640 hours, what is the MINIMUM SI flow required from Figure 1?
  2. Which ECCS Pumps can be stopped , if any, to reduce injection flow as close as possible to minimum flow determined from Item 1 (assume that ECCS Pump flows remain stable).

TASK STANDARD:

The task is satisfactorily met when the applicant has determined that the minimum flow per Figure 1 is 290 gpm (300 – 280 gpm acceptable band), and NONE of the running ECCS Pumps can be stopped.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

You are the CRS at Step 16.1 of 2-EOP-LOCA-5.

- DETERMINE the following:
  1. At Time 0640 hours, what is the MINIMUM SI flow required from Figure 1?

     290 gpm (300 – 280 gpm acceptable band)

  2. Which ECCS Pumps can be stopped , if any, to reduce injection flow as close as possible to minimum flow determined from Item 1 (assume that ECCS Pump flows remain stable).

     NONE of the ECCS Pumps can be stopped because neither an SI nor a Charging Pump alone can supply the minimum required flow injection of 290 gpm. Therefore, 21 SI and 22 Charging are required to remain in service.

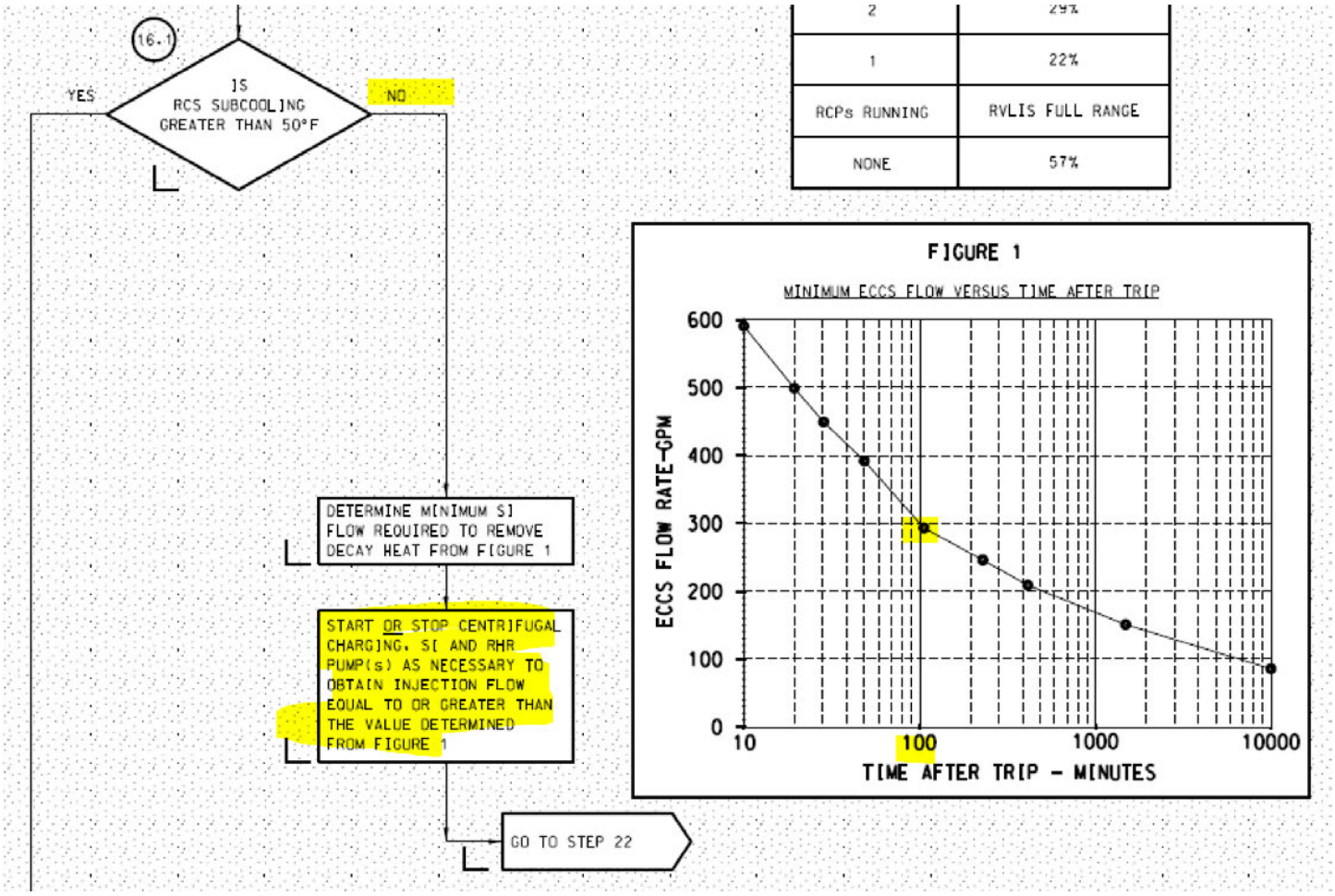
     Conditions from the stem:

     ECCS has been reduce to one train with the following ECCS Pump flow rates:

     - 21 SI Pump – 160 gpm
     - 22 Charging Pump – 220 gpm
     - BOTH RHR Pumps were stopped

ANSWER KEY (additional info)

From 2-EOP-LOCA-5, sheet 2 Rev. 41:



2	29%
1	22%
RCPs RUNNING	RVLIS FULL RANGE
NONE	57%

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate copy of 2-EOP-LOCA-5 sheet 2 **(D size sheet or greater**) Examiners Note: N/A Comments: N/A				
1.0	*	At Time 0640 hours, what is the MINIMUM SI flow required from Figure 1?	Determines that 100 minutes has elapsed since the Reactor Trip and using Figure 1 determines that 290 gpm (+/- 10 gpm) is the minimum ECCS flow injection.	
Cue: N/A **Examiners Note**: Figure 1 may be hard to read and readability errors will come into play. Lead Examiner will need to evaluate the applicants response to this answer to determine if the flow rate is reasonable based on the size of the figure. Comments:				
2.0	*	Which ECCS Pumps can be stopped ,if any, to reduce injection flow as close as possible to minimum flow determined from Item 1 (assume that ECCS Pump flows remain stable)	Determines that NONE of the ECCS Pumps can be stopped because neither an SI nor a Charging Pump alone can supply the minimum required flow injection of 290 gpm. Therefore, 21 SI and 22 Charging are required to remain in service	
Cue: N/A Examiners Note: N/A Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Terminating Cue		**JPM COMPLETE when applicant submits the worksheet.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC SRO-A1.a **Revision No.:** 0

Task Title: Determine Minimum ECCS Injection Flow

Task No.: N1150680502

Source:
Bank _____ New _____ Mod X

System: Conduct of Operations (Generic)

K/A Number / Description: G2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation

K/A Rating RO 4.4 SRO 4.7

Task Applicability: SRO Only X RO/SRO _____ AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **5** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: 2-EOP-LOCA-5 Sheet 2 (D size sheet or greater)

Reference(s): 2-EOP-LOCA-5

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT/WORKSHEET

Operator's Name: _____

INITIAL CONDITIONS

- At Time 0500, Salem Unit 2 was manually tripped and Safety Injection automatically actuated as a result of a LOCA
- At Time 0600, the RWST Low Level Alarm actuates and the crew transitions to 2-EOP-LOCA-3.
- The crew is unable to transfer to Cold Leg Recirculation and transitions to 2-EOP-LOCA-5, Loss of Emergency Recirculation, due to NO recirculation capability.
- ECCS has been reduce to one train with the following ECCS Pump flow rates:
  - 21 SI Pump – 160 gpm
  - 22 Charging Pump – 220 gpm
  - BOTH RHR Pumps were stopped

INITIATING CUE

You are the CRS at Step 16.1 of 2-EOP-LOCA-5.

- DETERMINE the following:
  1. At Time 0640 hours, what is the MINIMUM SI flow required from Figure 1?

  2. Which ECCS Pumps can be stopped , if any, to reduce injection flow as close as possible to minimum flow determined from Item 1 (assume that ECCS Pump flows remain stable).

RECORD YOU'RE ANSWERS ON THIS WORKSHEET

PSEG Nuclear LLC

Job Performance Measure

APPLY OVERTIME LIMIT RESTRICTIONS

JPM Number: 21-01 NRC SRO-A1.b

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**T. Shetrone /** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**M. Winkelspecth** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	New JPM for Salem. Used bank JPM from Diablo Canyon 2020 NRC NRCL181-A5. Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) LS-AA-119-1003 Rev. 10 Date Checked: 8/18/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

N/A 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/18/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. Copies of LS-AA-119-1003

INITIAL CONDITIONS

- It is 0600 on Monday 2/17
- Salem Unit 1 is at 100% power
- Salem Unit 2 is in an refueling outage and currently defueled
- The oncoming Unit 1 RO has just called in sick
- The Fatigue Rule Management System or EmpCenter is unavailable
- The following licensed operators are currently available **(see Next Page**):

INITIATING CUE

The Shift Manager directs you to perform the following:

- Review the hours worked by the available Operators and determine their ability to fill the watch bill. Document the reason(s) each operator is unavailable, if any. List of available Operators on **Next Page**.

Lists of Available Licensed Operators

Operator #1			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	0700	1900	RO
Monday, 2/10	0700	1900	RO
Tuesday, 2/11	0700	1900	BOP
Wednesday	0900	2200	Work Control NCO
Thursday, 2/13	Day Off		
Friday, 2/14	0700	2000	BOP
Saturday, 2/15	1400	1500	Telecon Personnel Statement
Sunday, 2/16	0700	1900	BOP
Monday, 2/17 (proposed)	0700	1900	RO

Operator #2			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	1900	0700	Work Control NCO
Monday, 2/10	1900	0700	Work Control NCO
Tuesday, 2/11	Day Off		
Wednesday	1900	0700	RO
Thursday, 2/13	1900	0700	Work Control NCO
Friday, 2/14	1900	0700	Work Control NCO
Saturday, 2/15	1900	0700	RO
Sunday, 2/16	Day Off		
Monday, 2/17 (proposed)	0700	1900	RO

Operator #3			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	1000	1700	Work Control NCO
Monday, 2/10	Day Off		
Tuesday, 2/11	0700	2100	BOP
Wednesday	0700	1900	BOP
Thursday, 2/13	0700	1900	RO
Friday, 2/14	0700	2000	BOP
Saturday, 2/15	1000	2000	RO
Sunday, 2/16	Day Off		
Monday, 2/17 (proposed)	0700	1900	RO

TASK STANDARD:

The task is satisfactorily met when the applicant has; (1) determined that Operators 1 and 3 DO NOT meet the work hour rule requirements of LS-AA-119-1003, Calculating Work Hours, AND (2) determines that Operator 2 does MEET the work hour rule requirements and available to take the watch on Monday 2/7. All documentation recorded to the reason why the operator is NOT available is in accordance with the attached answer key.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

Only Operator #2 can accept the watch without a waiver on the allowable work hours during Non-Outage Conditions.

Operator	Available to Fill the Watch (Circle one)	Why NOT selected
1	YES /NO	▪ Cannot exceed 16 hours in any continuous 24 period: **Satisfactory (max was 13 hours)** ▪ Cannot exceed 26 hours in any continuous 48 period: **Satisfactory (max was 25 hours)** ▪ Cannot exceed 72 hours in any continuous 168 period: **Satisfactory (max was 63 hours)** ▪ **34 hour continuous break required in any 9 day period: Unsatisfactory (taking the relief watch at 0700 on Monday would result in a continuous break of 33 hours in the past 9 day period)** ▪ 10 hour continuous break minimum break required between shifts: **Satisfactory (min was 12 hours)**
2	YES /NO	▪ Cannot exceed 16 hours in any continuous 24 period: **Satisfactory (max was 12 hours)** ▪ Cannot exceed 26 hours in any continuous 48 period: **Satisfactory (max was 24 hours)** ▪ Cannot exceed 72 hours in any continuous 168 period: **Satisfactory (max was 72 hours)** ▪ 34 hour continuous break required in any 9 day period: **Satisfactory (min was 36 hours)** ▪ 10 hour continuous break minimum break required between shifts: **Satisfactory (min was 12 hours)**
3	YES /NO	▪ Cannot exceed 16 hours in any continuous 24 period: **Satisfactory (max was 14 hours)** ▪ Cannot exceed 26 hours in any continuous 48 period: **Satisfactory (max was 24 hours)** ▪ **Cannot exceed 72 hours in any continuous 168 period: Unsatisfactory (max was 73.0 hours)** ▪ 34 hour continuous break required in any 9 day period: **Satisfactory(max was 38 hours)** ▪ 10 hour continuous break minimum break required between shifts: **Satisfactory (min was 10 hours)**

Additional Reference(s): LS-AA-119-1003, Calculating Work Hours

4.1.1. The following limits apply to covered individuals regardless of unit status:

- No more than 16 work hours in any 24-hour period.
- No more than 26 work hours in any 48-hour period.
- No more than 72 work hours in any 7-day period.

NOTE
FRMS conservatively uses 168 hours vs. 7-calendar days for this calculation and may produce deviations that may be overridden following verification that the 7-calendar day limit has not been exceeded.

- At least a 10-hour break between successive work periods or an 8-hour break when a break of less than 10 hours is necessary to accommodate a crew's scheduled transition between work schedules or shifts.
- A 34-hour break in any 9-calendar day period (this limit may be incorporated into the following table of limits).

LS-AA-119-1003 Page 28 of 37
CALCULATING WORK HOURS Rev: 10

Attachment 1, Examples

1. John Doe normally works a 12-hour shift schedule. He is requested to perform covered work for additional hours from 0700 to 1900 on Friday. What needs to be considered prior to working those additional hours?

   1.1. The following five items need to be considered:

   1.1.1. Review all hours worked during the 24-hour period prior to the stop time on Friday as reflected in the request to work additional hours to ensure no more than 16 hours in any 24-hour period will be worked.

   1.1.2. Review all hours worked during the 48-hour period prior to the stop time on Friday as reflected in the request to work additional hours to ensure no more than 26 hours in any 48-hour period will be worked.

   1.1.3. Review all hours worked during the 7-day period prior to the stop time on Friday (i.e., T-168 hours) as reflected in the request to work additional hours to ensure no more than 72 hours in any 7-day period will be worked.

   1.1.4. Review the break period(s) between the last day of work before Friday and 0700 on Friday to ensure a 10-hour break has been taken.

   1.1.5. Review the break period(s) during the last nine days prior to the stop time on Friday to ensure a break of at least 34 consecutive hours has been taken.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate copy of LS-AA-119-1003 Examiners Note: N/A Comments: N/A				
1.0	*	Determine if **Operator 1** can be selected.	Documented Operator 1 will violate the work hour rule requirement in LS-AA-119-1003 for not having a 34 hour break in any 9 calendar day period. Operator 1 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
2.0	*	Determine if **Operator 2** can be selected.	Documented Operator 2 MEETS the work hour rule requirements per LS-AA-119-1003. Operator 2 is AVAILABLE to take the watch.	
Cue: N/A Examiners Note: N/A Comments:				
3.0	*	Determine if **Operator 3** can be selected.	Documented Operator 3 will violate the work hour rule requirement in LS-AA-119-1003 for working more than 72 hours in a 7 day period. Operator 2 is ELIMINATED.	
Cue: N/A Examiners Note: N/A				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
Terminating Cue		**JPM COMPLETE when the applicant submits the worksheet.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC SRO-A1.b **Revision No.:** 0

Task Title: Apply Overtime Limit Restrictions

Task No.: N1230040302

Source: Bank X New _____ Mod _____

System: Conduct of Operations (Generic)

K/A Number / Description: G2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential).

K/A Rating RO 2.9 SRO 3.9

Task Applicability: SRO Only X RO/SRO _____ AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **30** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: LS-AA-119-1003

Reference(s): LS-AA-119-1003

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- It is 0600 on Monday 2/17
- Salem Unit 1 is at 100% power
- Salem Unit 2 is in an refueling outage and currently defueled
- The oncoming Unit 1 RO has just called in sick
- The Fatigue Rule Management System or EmpCenter is unavailable
- The following licensed operators are currently available (see Next Page):

INITIATING CUE

The Shift Manager directs you to perform the following:

- Review the hours worked by the available Operators and determine their ability to fill the watch bill. Document the reason(s) each operator is unavailable, if any.
- List of available Operators on Next Page.
- Document your selection and reason on the attached worksheet.

Lists of Available Licensed Operators

Operator #1			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	0700	1900	RO
Monday, 2/10	0700	1900	RO
Tuesday, 2/11	0700	1900	BOP
Wednesday	0900	2200	Work Control NCO
Thursday, 2/13	Day Off		
Friday, 2/14	0700	2000	BOP
Saturday, 2/15	1400	1500	Telecon Personnel Statement
Sunday, 2/16	0700	1900	BOP
Monday, 2/17 (proposed)	0700	1900	RO

Operator #2			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	1900	0700	Work Control NCO
Monday, 2/10	1900	0700	Work Control NCO
Tuesday, 2/11	Day Off		
Wednesday	1900	0700	RO
Thursday, 2/13	1900	0700	Work Control NCO
Friday, 2/14	1900	0700	Work Control NCO
Saturday, 2/15	1900	0700	RO
Sunday, 2/16	Day Off		
Monday, 2/17 (proposed)	0700	1900	RO

Operator #3			
Day	**Start Time**	**End Time**	**Position**
Sunday, 2/9	1000	1700	Work Control NCO
Monday, 2/10	Day Off		
Tuesday, 2/11	0700	2100	BOP
Wednesday	0700	1900	BOP
Thursday, 2/13	0700	1900	RO
Friday, 2/14	0700	2000	BOP
Saturday, 2/15	1000	2000	RO
Sunday, 2/16	Day Off		
Monday, 2/17 (proposed)	0700	1900	RO

STUDENT WORKSHEET

Operator's Name: _____

Operator	Available to Fill the Watch (Circle one)	Why NOT selected
1	YES /NO	
2	YES /NO	
3	YES /NO	

PSEG Nuclear LLC

Job Performance Measure

REVIEW A COMPLETED SURVEILLANCE TEST

JPM Number: 21-01 NRC SRO-A2

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**T. Shetrone** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	Converted to new template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-ST.CBV-0001 Rev. 9
S2.RA-ST.CBV-0001 Rev. 7
Date Checked: 7/25/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11/28/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. Copies of Unit 2 Tech Specs for each applicant.

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- The BOP operator just completed performing S2.OP-ST.CBV-0001, Inservice Testing Containment Ventilation Valves Modes 1-6.

INITIATING CUE

You are the CRS, complete the review of the completed surveillance procedure for accuracy and completeness starting at Step 5.4.3.A.

Document any discrepancies and evaluate Tech Specs if necessary.

TASK STANDARD:

The task is satisfactorily met when the applicant has determined the 2VC13 stroke time is in the Required Action Range and as a minimum TS LCO 3.6.3 Action 1.a is applicable.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional)

S2.OP-ST.CBV-0001(Q)

ATTACHMENT 2
(Page 1 of 2)

STROKE TIME DATA

1.0 **CONTAINMENT VENTILATION SAMPLE, REGULAR AND BACKUP, ISOLATION VALVES OPERATED FROM 2RP1:**

Component Description	Stroke	Stroke Time (sec)	Stopwatch Number	Stroke Date/ Time	Initials	Test Results		
						Acceptable Range SAT	Required Evaluation Range SAT	Required Action Range UNSAT
2VC7 21R Outlet VA	X (1) (3)	1.41	220940	Today	JD	JD		
2VC11 21R Inlet VA	X (1) (3)	1.88	220945	Today	JD	JD		
2VC8 22R Outlet VA	X (1) (3)	1.33	220940	Today	JD	JD		
2VC12 22R Inlet VA	X (1) (3)	1.58	220945	Today	JD	JD		
2VC9 21B Outlet VA	X (2) (3)	1.09	220940	Today	JD	JD		
2VC13 21B Inlet VA	X (2) (3)	2.83	220945	Today	JD	JD		
2VC10 22B Outlet VA	X (2) (3)	1.33	220940	Today	JD	JD		
2VC14 22B Inlet VA	X (2) (3)	1.76	220945	Today	JD	JD		
Acceptance Criteria:		Measured values are within bands specified in S2.RA-ST.CBV-0001(Q), Inservice Testing Containment Ventilation Valves Modes 1-6 Acceptance Criteria OR data represents new baseline data as determined by the IST Implementation Engineer.						

(1) **ENSURE** Containment BU Isolation Valves, 2VC9/13 and 2VC10/14, are OPEN.

(2) **ENSURE** Containment REG Isolation Valves, 2VC7/11 and 2VC8/12, are OPEN.

(3) Back Up Isolation Valves must be manually CLOSED from 2RP1 if a Phase A Containment Isolation Signal occurs.

Additional Reference(s):

S2.RA-ST.CBV-0001(Q) Page 7 of 8

INSERVICE TESTING CONTAINMENT VENTILATION VALVES MODES 1-6 ACCEPTANCE CRITERIA Rev: 7

Attachment 1, STROKE TIME DATA ACCEPTANCE CRITERIA

1.0 VALVE PERFORMANCE DATA:

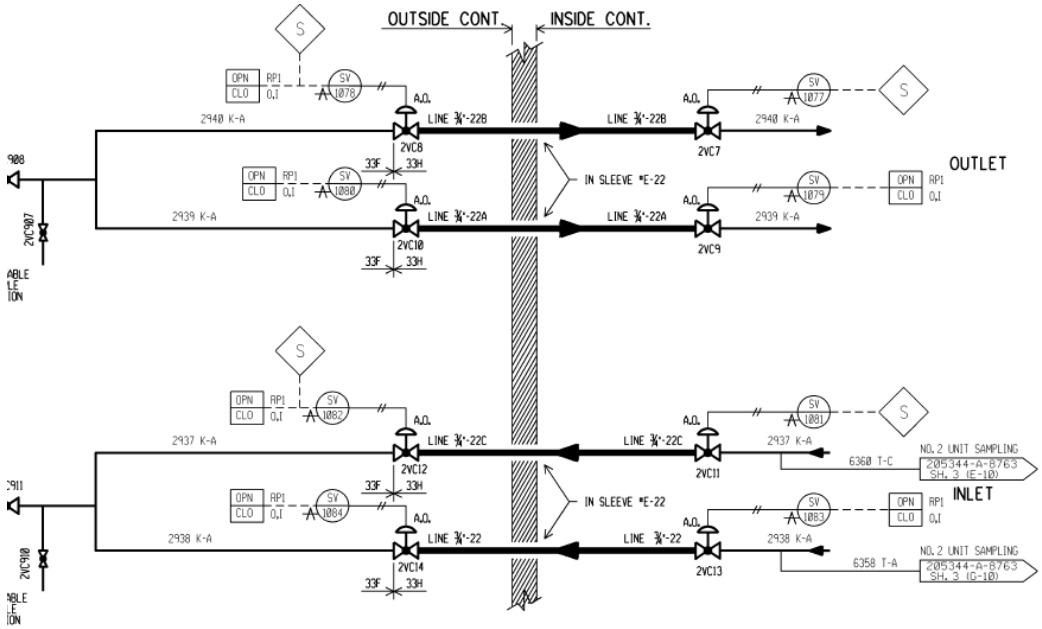
Component	Stroke	Reference Value (seconds)	Date	Test Results					
				Acceptable Range (seconds)	Required Evaluation (seconds)		Required Action Range (seconds)		TS/FSAR (seconds)
					Low	High	Low	High	
2VC7	X (1)	1.5 (2)	11/21/95	≤ 2.0	N/A	N/A	N/A	> 2.0	≤ 10.0
2VC11	X (1)	1.1 (2)	11/21/95	≤ 2.0	N/A	N/A	N/A	> 2.0	≤ 10.0
2VC8	X (1)	1.5 (6)	06/19/2001	0.8 - 2.3	< 0.8	> 2.3 - 3.0	N/A	> 3.0	≤ 10.0
2VC12	X (1)	1.8 (5)	10/14/00	0.9 - 2.7	< 0.9	> 2.7 - 3.6	N/A	> 3.6	≤ 10.0
2VC9	X (1)	0.9 (2)	11/21/95	≤ 2.0	N/A	N/A	N/A	> 2.0	N/A
2VC13	X (1)	0.9 (2)	11/21/95	≤ 2.0	N/A	N/A	N/A	> 2.0	N/A
2VC10	X (1)	1.9 (2)	01/27/2011	1.0 - 2.9	<1.0	>2.9 - 3.8	N/A	> 3.8	N/A
2VC14	X (1)	1.7 (2)	05/06/2009	0.9 - 2.6	<0.9	>2.6 - 3.4	N/A	> 3.4	N/A

(1) Credit is taken for satisfying the fail safe position.

(2) Reference value based on Order 40026547.

(5) Reference value based on Order 50027557.

(6) Reference value based on Order 60020334.



CONTAINMENT RADIATION MONITORING SYSTEM
SAMPLE INLET & OUTLET ISOLATION VALVES
REFERENCE DRAWING 238805-B-9638 (PIPING)

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate completed surveillance procedure (S2.OP-ST.CBV-0001), S2.RA-ST.CBV-0001 **and** Unit 2 Tech Specs. Examiners Note: N/A Comments: N/A				
5.4.3.A		**REVIEW** this procedure with Attachments 1-4 for completeness and accuracy.	Applicant determines that 2VC13 stroke time was incorrectly recorded as SAT and should be "REQUIRED ACTION RANGE"	
Cue: N/A Examiners Note: N/A Comments:				
5.4.3.B		IF valve stroke times are within the ACCEPTABLE RANGE, THEN **DECLARE** the applicable valve(s) OPERABLE.	Applicant determines that 2VC13 is INOPERABLE.	
Cue: N/A Examiners Note: N/A Comments:				
5.4.3.C		IF ANY valve's stroke time is within the REQUIRED EVALUATION RANGE, THEN:	Determines N/A	
Cue: N/A Examiners Note: N/A Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.4.3.D.1	*	IF ANY valve's stroke time is in the REQUIRED ACTION RANGE, THEN: Immediately **DECLARE** the respective valve(s) inoperable.	Applicant determines 2VC13 INOPERABLE based on the stroke time in the REQUIRED ACTION RANGE.	
Cue: N/A Examiners Note: N/A Comments:				
5.4.3.D.2	*	**EVALUATE** Technical Specification requirements for system operability.	Applicant determine that TS LCO 3.6.3 is NOT MET for 2VC13. TS action 1a – restore the inoperable valve(s) to operable status within 4 hours, OR TS action 1.b the 2VC14 must be closed and deactivated within four hours of the stroke time. OR TS action 1.c the penetration must be isolated within four hours of the stroke time. OR TS action 1.d - HOT STANDBY in the next 6 hours and COLD SHUTDOWN in the following 30 hours.	
Cue: N/A Examiners Note: N/A Comments:				
Terminating Cue		**JPM COMPLETE when the applicant submits the worksheet.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC SRO-A2 **Revision No.:** 0

Task Title: Review a Completed Surveillance Test

Task No.: N1230010302 **Source:** Bank X New _____ Mod _____

System: Equipment Control (Generic)

K/A Number / Description: G2.2.12 Knowledge of surveillance procedures.

K/A Rating RO 3.7 SRO 4.1

Task Applicability: SRO Only X RO/SRO _____ AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: S2.OP-ST.CBV-0001, S2.RA-ST.CBV-0001, and Salem Unit 2 Tech Specs

Reference(s): S2.OP-ST.CBV-0001 S2.RA-ST.CBV-0001 Salem Unit 2 Tech Specs

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT/WORKSHEET

Operator's Name: _____

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- The BOP operator just completed performing S2.OP-ST.CBV-0001, Inservice Testing Containment Ventilation Valves Modes 1-6.

INITIATING CUE

You are the CRS, review the completed surveillance procedure for accuracy and completeness starting at Step 5.4.3.A.

Document any discrepancies and evaluate Tech Specs if necessary.

RECORD YOU'RE RESPONSE ON THIS WORKSHEET

PSEG Nuclear LLC

Job Performance Measure

SELECT A VOLUNTEER FOR EMERGENCY RADIATION EXPOSURE

JPM Number: 21-01 NRC SRO-A3

Revision Number: 0

Date: 08/18/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	8/18/22
Validated By:	**T. Shetrone /** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	8/18/22	New JPM for Salem. Used bank JPM from Commanche Peak 2019 NRC SA4. Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

RC Procedure(s) NC.EP-EP.ZZ-0304 Rev. 17 Date Checked: 8/18/22
NC.EP-EP.ZZ-0102 Rev. 27

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

N/A 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/18/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. Copies of NC.EP-EP.ZZ-0304 for each applicant.

INITIAL CONDITIONS

- Salem Unit 1 is experiencing a LBLOCA
- Transfer to Cold Leg Recirculation has been established
- An operator venting 11 RHR Pump is injured and needs assistance to exit the RCA
- Emergency Coordinator (EC) authorized one volunteer to attempt lifesaving task
- Rad Pro predicts dose of 15-20 REM given general radiation levels in the area of the injured person

INITIATING CUE

The EC directs you to perform the following:

- Evaluate and select the most preferred volunteer from the list of available volunteers per NC.EP-EP.ZZ-0304, Operational Support Center (OSC) Radiation Protection Response.

List of Available Volunteers

Volunteer 1:	46 year old male, newly hired Custodian. No prior radiation protection training.
Volunteer 2:	27 year old female; informs you she is pregnant.
Volunteer 3:	24 year old male, recently qualified NEO.
Volunteer 4:	52 year old male, lifetime exposure 150 REM, including a Protecting Valuable Property dose of 15 REM at another facility.
Volunteer 5:	49 year old male, works in Maintenance; informs you that he is on prescription medication that may cause dizziness.
Volunteer 6:	42 year old female, not capable of becoming pregnant.
Volunteer 7:	50 year old male, SRO at Salem. Recently removed from License duties due to a failed periodic written exam.

TASK STANDARD:

The task is satisfactorily met when the applicant has; (1) determined that Volunteers 1-6 did NOT meet the requirements of NC.EP-EP.ZZ-0304, AND (2) determines that Volunteer 7 does MEET the requirements of NC.EP-EP.ZZ-0304 to perform lifesaving tasks. All documentation recorded to the reason why or why NOT is in accordance with the attached answer key.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

Volunteer	Selected (Circle one)	Why or Why NOT selected
1	NO	Determined individual NOT familiar with consequences of exposure to radiation.
2	NO	Determined female volunteer has declared pregnancy.
3	NO	Determined individual has NOT reached age of 45 years or older.
4	NO	Determined individual already received emergency exposure for Protecting Valuable Property.
5	NO	Determined individual is NOT physically qualified based on prescription medication that causes dizziness.
6	NO	Determined individual has NOT reached age of 45 years of older.
7	YES	Selected Volunteer 7 as most preferred candidate for Emergency exposure because he is over the age of 45, has not received a previous emergency exposure in his lifetime, he is familiar with the task and consequences of radiation exposure as a licensed operator, and he is not a female capable of reproduction

Additional References:

NC.EP-EP.ZZ-0102:

	~~Station has adequate resources.~~				
SM	Until relieved by the EDO, as needed/requested for accident mitigation (25R) or life saving (75R), **AUTHORIZE** Emergency Dose. *(per OSC RP Response, NC.EP-EP.ZZ-0304)* • **ENSURE** RAC works with the OSC and considers dose savings to the public to expedite OSC team dispatch. • **ENSURE** backup actions are being planned; (OSC or TSC engineering staff). • **ENSURE** OSC Coordinator considers pre-staging teams at the control point while plans are being finalized.	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____	Init:____ Time:____

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate copy of NC.EP-EP.ZZ-0304 Examiners Note: N/A Comments: N/A				
1.0	*	Determine if **Volunteer 1** can be selected.	Documented Volunteer 1 is NOT familiar with the consequences of exposure to radiation and/or have attended and passed Rad Worker Training. Volunteer 1 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
2.0	*	Determine if **Volunteer 2** can be selected.	Documented Volunteer 2 is a declared pregnant women and SHALL NOT be allowed to volunteer. Volunteer 2 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
3.0	*	Determine if **Volunteer 3** can be selected.	Documented Volunteer 3 is NOT above the age of 45 years, although qualified for the task as an NEO. Volunteer 3 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
4.0	*	Determine if **Volunteer 4** can be selected.	Documented Volunteer 4 already received emergency exposure for Protecting Valuable Property. Volunteer 4 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
5.0	*	Determine if **Volunteer 5** can be selected.	Documented Volunteer 5 is NOT physically qualified due to taking prescription medication. Volunteer 5 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
6.0	*	Determine if **Volunteer 6** can be selected.	Documented Volunteer 6 is NOT above the age of 45 years. Volunteer 6 is ELIMINATED.	
Cue: N/A Examiners Note: N/A Comments:				
7.0	*	Determine if **Volunteer 7** can be selected.	SELECTED Volunteer 7 for emergency exposure and DOCUMENTED as the most preferred candidate because he is over the age of 45, has not received a previous emergency exposure in his lifetime, he is familiar with the task and consequences of radiation exposure as a	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
			licensed operator, and he is not a female capable of reproduction.	
Cue: N/A Examiners Note: N/A Comments:				
Terminating Cue	**JPM COMPLETE when the applicant submits the worksheet.**			N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC SRO-A3 **Revision No.:** 0

Task Title: Select a Volunteer for Emergency Radiation Exposure

Task No.:

Source:
Bank X New _____ Mod _____

System: Radiation Control (Generic)

K/A Number / Description: G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only).

K/A Rating RO N/A SRO 3.8

Task Applicability: SRO Only X RO/SRO _____ AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **20** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: NC.EP-EP.ZZ-0304

Reference(s): NC.EP-EP.ZZ-0102 NC.EP-EP.ZZ-0304

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Salem Unit 1 is experiencing a LBLOCA
- Transfer to Cold Leg Recirculation has been established
- An operator venting 11 RHR Pump is injured and needs assistance to exit the RCA
- Emergency Coordinator (EC) authorized one volunteer to attempt lifesaving task
- Rad Pro predicts dose of 15-20 REM given general radiation levels in the area of the injured person

INITIATING CUE

The EC directs you to perform the following:

- Evaluate and select the most preferred volunteer from the list of available volunteers per NC.EP-EP.ZZ-0304, Operational Support Center (OSC) Radiation Protection Response.

List of Available Volunteers

Volunteer 1:	46 year old male, newly hired Custodian. No prior radiation protection training.
Volunteer 2:	27 year old female; informs you she is pregnant.
Volunteer 3:	24 year old male, recently qualified NEO.
Volunteer 4:	52 year old male, lifetime exposure 150 REM, including a Protecting Valuable Property dose of 15 REM at another facility.
Volunteer 5:	49 year old male, works in Maintenance; informs you that he is on prescription medication that may cause dizziness.
Volunteer 6:	42 year old female, not capable of becoming pregnant.
Volunteer 7:	50 year old male, SRO at Salem. Recently removed from License duties due to a failed periodic written exam.

STUDENT WORKSHEET

Operator's Name: _____

Volunteer	Selected (Circle one)	Why or Why NOT selected
1	YES /NO	
2	YES /NO	
3	YES /NO	
4	YES /NO	
5	YES /NO	
6	YES /NO	
7	YES /NO	

PSEG Nuclear LLC

Job Performance Measure

REVIEW INITIAL CONTACT MESSAGE FORM (ICMF)

JPM Number: 21-01 NRC SRO-A4

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**T. Shetrone** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	New JPM. Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC	1. Task description and number, JPM description and number are identified.
RC	2. Knowledge and Abilities (K/A) references are included.
RC	3. Performance location specified. (in-plant, control room, or simulator)
RC	4. Initial setup conditions are identified.
RC	5. Initiating and terminating cues are properly identified.
RC	6. Task standards identified and verified by SME review.
RC	7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)
RC	8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:
RC	Procedure(s) EP-SA-325-F2 Rev. 1 Date Checked: 7/24/22
RC	9. Pilot test the JPM:
	a. Verify cues both verbal and visual are free of conflict, and
	b. Ensure performance time is accurate.
N/A	10. If the JPM cannot be performed as written with proper responses, then revise the JPM.
N/A	11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	8/18/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an Admin JPM.

SPECIAL INSTRUCTIONS

1. Copies of Salem ECG Flowchart and Bases documents for each SRO.
2. Marked up EP-SA-325-F2 Attachment 2 Alert (includes all pages of Attachment 2)

INITIAL CONDITIONS

- At 1055, Salem Unit 2 initiated a manual SI due to a Steam Generator Tube Rupture on 23 SG
- RCS temperature is 543 °F
- BOTH motor driven auxiliary feedpumps failed to start and the Turbine Driven auxiliary feedpump (23 AFW Pump) automatically started and is feeding all intact SGs at 22E4 lbm/hr
- At 1100, the Shift Manager declared an ALERT
- 33 Ft. level wind direction is from 180 degrees at 5 mph
- No additional operator actions have been taken

INITIATING CUE

The Shift Manager directs you to perform an accuracy peer check on the provided Initial Contact Message Form (ICMF).

Document any errors on the attached worksheet, should any exist.

This is a TIME CRITICAL JPM

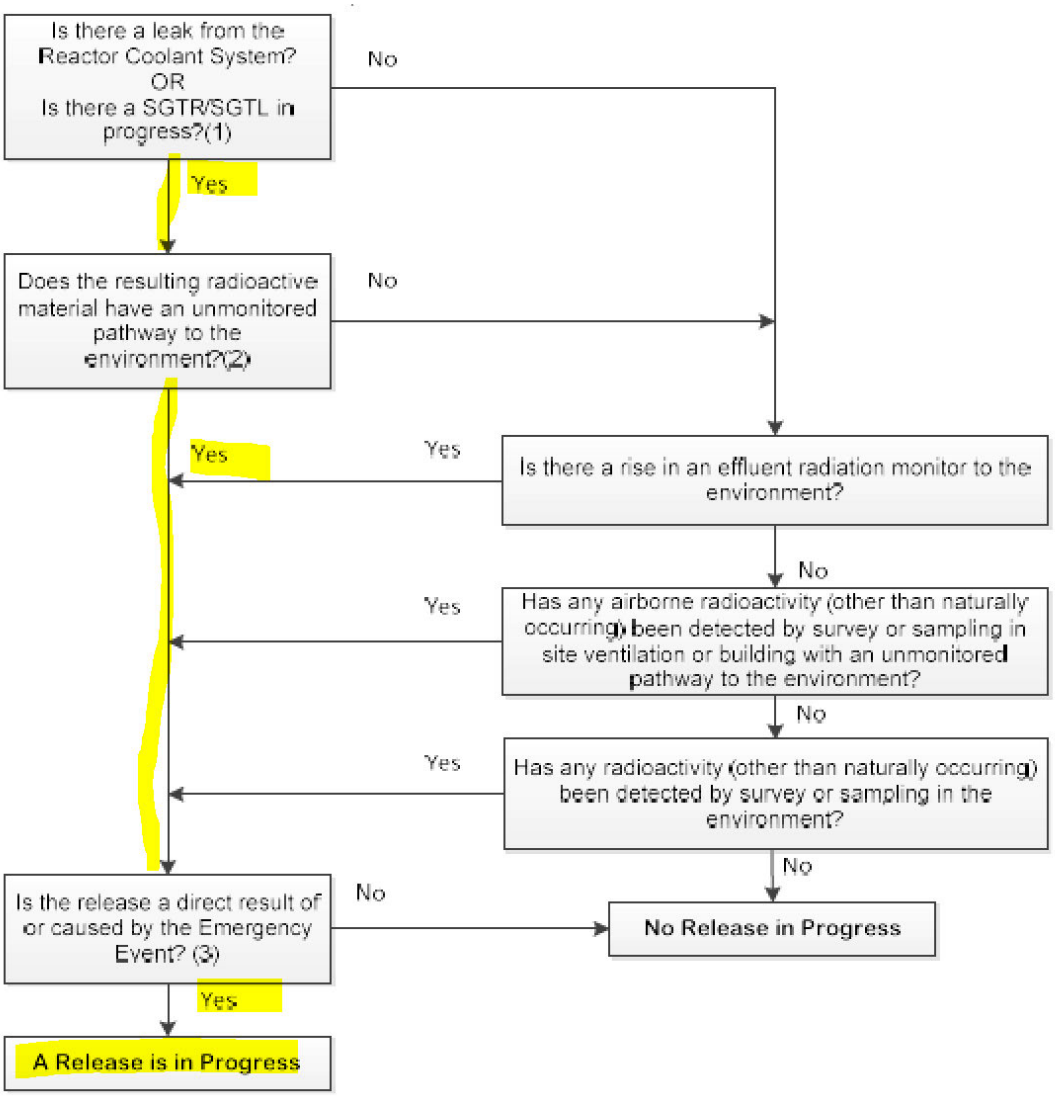
Additional Reference(s): EP-SA-325-F2 Rev. 01

- **Accuracy Peer Check:** Have the STA, TSS, SSM or other knowledgeable individual perform an accuracy peer check of the ICMF. The peer check should include:
  - verification that the correct form has been used for the classification
  - review of all applicable fields that have been completed by the SM during completion of the ICMF (date, time, EAL, Description etc.) to ensure data is correct/accurate
  - EC approval initials have been completed.

- **Classification Timeliness:** Classification must occur after verification and before the 15 minute EAL clock expires. Assessment of an Emergency Condition should be completed in a timely manner, which is considered to be within 15 minutes of when events are known or should have been known. If an EAL specifies a duration time (e.g., loss of annunciators for 15 minutes or longer), the assessment time runs concurrently with the EAL duration time and is the same length. [3]

- **Rad Release Determination (Salem):** Any of the following conditions constitutes a release in progress due to the event:
  - Increase in Plant Vent Release Rate (NOT caused by operational transients or ventilation changes) which is caused by the EVENT.
  - A Steam Generator rupture which is faulted to the atmosphere (SRV, Atmospheric Vent or Line Break)
  - Evidence of an UNMONITORED release to atmosphere, examples include:
    - Primary to Secondary leak and the steam driven feed pump is in service
    - Bypass release from the Aux Building through the Electrical Pen
    - Leakage from the Containment directly to atmosphere after a LOCA
    - Elevated background radiation levels downwind from the station as monitored/reported by Radiation Protection Technician at fence line

Additional Reference(s): EP-SA-325-F2 Release in Progress Determination Flowchart



Additional Reference(s): EP-SA-325-102 Rev. 1, page 7, Event Classification Guide Use

4.9 Identify and implement the referenced ECG form based on the Emergency Classification Level.

- Unusual Event — Implement EP-SA-325-F1
- Alert — Implement EP-SA-325-F2
- Site Area Emergency — Implement EP-SA-325-F3
- General Emergency — Implement EP-SA-325-F4
- Unusual Event (Common Site) — Implement EP-SA-325-F24

Refer to TABLE 1, Salem Emergency Classification Description Table for correct description wording to be used on the Initial Contact Message Form (ICMF) for all EALs.

TASK STANDARD:

The task is satisfactorily met when the applicant has identified the following errors on the ICMF; (1) declaration time is incorrect, (2) description of the event is not in accordance with Table 1 of EP-SA-325-102, and (3) a release not in progress is incorrectly checked off. The ICMF should be in accordance with the attached answer key. **This is a TIME CRITICAL JPM which must be completed ≤ 15 minutes.**

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

EP-SA-325-F2
ATT 2
Pg. 2 of 6

INITIAL CONTACT MESSAGE FORM

I. THIS IS John Q (NAME), COMMUNICATOR IN THE ☒ CONTROL ROOM ☐ TSC ☐ EOF

AT THE **SALEM** NUCLEAR GENERATING STATION, **UNIT**(s) No. 2

II. THIS IS NOTIFICATION OF AN **ALERT** WHICH WAS

DECLARED AT 1100 (Time - 24 HR CLOCK) ON Today (DATE)

EAL # RB1.L DESCRIPTION OF EVENT

Loss of the Reactor Coolant System Barrier

III. ☒ THERE **IS** A RELEASE IN PROGRESS DUE TO THE EVENT
☐ THERE IS **NO** RELEASE IN PROGRESS DUE TO THE EVENT

} Any release above normal, attributable to the event. See Basis for examples.

IV. 33 FT. LEVEL WIND DIRECTION (**From**): 180 (DEGREES) WIND SPEED: 5 (MPH)
(From MET Computer /SPDS)

V. **NO PROTECTIVE ACTIONS ARE RECOMMENDED AT THIS TIME**

Shift Manager / SM
EC Print/Sign
(Approval to Transmit ICMF)

Answer Key

ICMF Section	Errors Identified (Circle one)	Brief Description of Error(s)
I	NO	
II	YES	1. Declaration time is incorrect should be 1100, 2. Description of Event is incorrect should read "Loss of the Reactor Coolant System Barrier" per Table 1 of EP-SA-325-102
III	YES	1. A release is in progress due to the Turbine Driven auxiliary feedpump the ONLY pump in service and supplied from 21 and 23 SGs. No actions have been taken yet to close the 23MS45 steam supply from 23 SG. Per EP-SA-325-F2, a release is in progress due to a Primary to Secondary leak and the steam driven feedpump in service with no radiation monitor to monitor the release path.
IV	NO	
V	NO	

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate MARKED UP copy of EP-SA-325-F2 Attachment 2 Alert(include all pages from Attachment 2) **and** Salem ECG Flowcharts and Bases Books. Examiners Note: N/A Comments: N/A				
1.0	*	Determine if **Section I** has any errors.	No errors	
Cue: N/A Examiners Note: N/A Comments:				
2.0	*	Determine if **Section II** has any errors.	Documents that Section II has two errors: (1) declaration time is incorrect should be 1100, and (2) the Description of Event is incorrect should read "Loss of the Reactor Coolant System Barrier".	
Cue: N/A Examiners Note: N/A Comments:				
3.0	*	Determine if **Section III** has any errors.	Documents that Section III incorrectly checked that NO release is in progress. A primary to secondary leak and the steam driven feedpump in service is a release in progress.	
Cue: N/A Examiners Note: N/A				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
4.0	*	Determine if **Section IV** has any errors.	No errors	
Cue: N/A Examiners Note: N/A Comments:				
5.0	*	Determine if **Section V** has any errors.	No errors	
Cue: N/A Examiners Note: N/A Comments:				
Terminating Cue		**JPM COMPLETE when the applicant submits the worksheet.**		N/A

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC SRO-A4 **Revision No.:** 0

Task Title: Review Initial Contact Message Form (ICMF)

Task No.: N1240010502 **Source:** Bank _____ New X Mod _____

System: Emergency Procedures/Plan (Generic)

K/A Number / Description: G2.4.40 Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only).

K/A Rating RO N/A SRO 4.5

Task Applicability: SRO Only X RO/SRO _____ AO/RO/SRO _____ Other _____

Time-Critical: Yes X No _____ **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **15** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom X Simulator _____ In-Plant _____ RCA _____

Required Materials: Salem ECG & Basis Document

Reference(s): Salem ECG

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- At 1055, Salem Unit 2 initiated a manual SI due to a Steam Generator Tube Rupture on 23 SG
- RCS temperature is 543 °F
- BOTH motor driven auxiliary feedpumps failed to start and the turbine driven auxiliary feedpump (23 AFW Pump) automatically started and is feeding all intact SGs at 22E4 lbm/hr
- At 1100, the Shift Manager declared an ALERT
- 33 Ft. level wind direction is from 180 degrees at 5 mph
- No additional operator actions have been taken

INITIATING CUE

The Shift Manager directs you to perform an accuracy peer check on the provided Initial Contact Message Form (ICMF).

Document any errors on the attached worksheet, should any exist.

This is a TIME CRITICAL JPM

STUDENT WORKSHEET

Operator's Name: _____

ICMF Section	Errors Identified (Circle one)	Brief Description of Error(s)
I	YES /NO	
II	YES /NO	
III	YES /NO	
IV	YES / NO	
V	YES / NO	

PSEG Nuclear LLC

Job Performance Measure

Start the SBO Air Compressor IAW AB.LOOP-1

JPM Number: 21-01 NRC IP-i

Revision Number: 1

Date: 10/14/2022

Developed By:	**K. Hantho** **Instructor**	**Date:**	10/14/22
Validated By:	**N/A (see revision comment)** **SME or Instructor**	**Date:**	11/28/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
01	11/28/22	Converted to new JPM template. No validation required since previously used during annual exams and procedure has not changed since last use.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

KH 1. Task description and number, JPM description and number are identified.

KH 2. Knowledge and Abilities (K/A) references are included.

KH 3. Performance location specified. (in-plant, control room, or simulator)

KH 4. Initial setup conditions are identified.

KH 5. Initiating and terminating cues are properly identified.

KH 6. Task standards identified and verified by SME review.

KH 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

KH 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-AB.CR-0002 Rev. 34 Date Checked: 8/16/22

9. Pilot test the JPM:

   a. Verify cues both verbal and visual are free of conflict, and

   b. Ensure performance time is accurate.

10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

11. When JPM is revalidated, SME or instructor sign and date JPM cover page

K. Hantho	10-14-22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an In-plant JPM.

SPECIAL INSTRUCTIONS

- **This JPM is NOT located inside the RCA.**
- **Need L-3 key to unlock SBO Compressor Building**

INITIAL CONDITIONS

- Both Salem Units have tripped due to a loss of off-site power. Equipment problems have resulted in NO Emergency Control Air Compressors (ECACs) running.

INITIATING CUE

- The Unit 2 CRS has directed you to start the Station Blackout Compressor (SBO) IAW SC.OP-SO.CA-0001, SBO Diesel Control Air Compressor, starting with Step 5.1.3.
- The Component Off Normal and Off Normal Tagged review has been performed SAT.
- It is **NOT** the summer operating period.
- The inspection covers have already been removed, Step 5.1.1 was performed SAT, and the inspection covers replaced IAW Step 5.1.2

TASK STANDARD:

The task is satisfactorily met when the applicant starts the SBO Diesel Control Air Compressor, aligns the discharge to the air headers, and loads the compressor.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: **READ TO APPLICANT:** No plant controls or equipment are to be operated during the performance of this JPM. All actions taken by the examinee should be clearly demonstrated and verbalized to the evaluator. **Examiners Note**: The candidate may wish to review all P&Ls prior to start. Comments: N/A				
N/A		Provide marked up copy of the SC.OP-SO.CA-0001, SBO Diesel Control Air Compressor with Prerequisites signed off. Additionally, Step 5.1.1 for removing Inspection Covers will be previously completed, as will Step 5.1.2 to re-install inspection covers.	Operator reviews and signs Precautions and Limitations. Operator must address obtaining hearing protection per P&L 3.1 All other P&Ls require no action other than reading and initialing with the exception of P&L 3.10 is verified at the SBO.	N/A
Cue: Examiners Note: None Comments:				
5.1.3		**OPEN** both Engine intake louvers (located outside building on west wall.)	**Locates louvers and demonstrates how to open.**	
Cue: After locating and describing how to open Engine intake louvers, state that the Engine intake louvers are open. Examiners Note: None Comments:				
5.1.4		**OPEN** 1CA1920, BLACKOUT AIR COMPRESSOR DRAIN VALVE, to drain any water accumulation, then CLOSE 1CA1920.	**Since the cue would be the same whether or not the drain cap was removed (nothing came out the end of**	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
			pipe) only comment here for review if pipe cap was not removed.	
Cue: <u>After</u> locating 1CA1920, BLACKOUT AIR COMPRESSOR DRAIN VALVE, and describing how to open, including that the drain cap must be removed, state 1CA1920, BLACKOUT AIR COMPRESSOR DRAIN VALVE, has been cycled and no water came out of pipe. Examiners Note: None <u>Comments:</u>				
5.1.5		**OPEN** 2FZSBO10, BATTERY CHARGER, breaker.	Locates 2FZSBO10, BATTERY CHARGER, and simulates opening.	
Cue: 2FZSBO10, BATTERY CHARGER, breaker, is open. **Examiners Note: breaker in Panel 2FZSBO, SBO Compressor Bldg Distribution Panel** <u>Comments:</u>				
5.1.6		**PLACE** SBO Control Air Dryer switch in ON position.	Locates SBO Control Air Dryer switch and simulates turning switch to ON position.	
Cue: **SBO Control Air Dryer switch is ON, and Control Air On light is illuminated.** Examiners Note: <u>Comments:</u>				
5.1.7		**RECORD** Engine Hourmeter reading on Attachment 1, Section 1.0, and in log book at SBO compressor.	Locates Engine Hourmeter, 1XA16993, Tachometer/Hourmeter, inside engine control panel labeled Compressor Noise Emission Control Information, and records current reading in Section 1.0 of Att. 1. Locates logbook hanging from side of panel 2FZSBO, SBO Compressor Bldg Distribution Panel, and identifies where Engine Hourmeter should be logged.	
Cue:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Examiners Note: Comments:				
5.1.8		**PLACE** UNLOADER VALVE selector switch in START position.	Locates UNLOADER VALVE selector switch, and ensures switch in START position.	
Cue: None **Examiners Note**: **UNLOADER VALVE selector switch is normally in Start when engine is shutdown.** Comments:				
5.1.9	*	**PRESS** AND **HOLD** BY-PASS VALVE pushbutton.	**Locates BY-PASS VALVE pushbutton in control panel, and simulates depressing AND holding depressed.**	
Cue: None Examiners Note: None Comments:				
5.1.10	*	**PLACE** SBO-IGN-SWT, ENGINE IGNITION SWITCH in START position until engine starts, then **RELEASE** to RUN position.	**While keeping the Bypass Valve pushbutton depressed, simulates placing the SBO-IGN-SWT SBO Air Comp Inst Pnl Engine Ignition Switch in START.** **After cue is given, releases SBO-IGN-SWT SBO Air Comp Inst Pnl Engine Ignition Switch to RUN.**	
Cue: Engine has started. **Examiners Note:** If also releases the BYPASS VALVE pushbutton at this point, without checking oil pressure >15 psig, then CUE that the engine has stopped. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.1.11	*	When engine oil pressure is >15 psig, RELEASE BY-PASS VALVE pushbutton.	**Identifies oil pressure reading on gauge labeled" oil pressure" gauge inside control panel.** **After cue releases BY-PASS VALVE pushbutton.**	
Cue: Oil pressure is 40 psig. Examiners Note: None Comments:				
5.1.12		**ALLOW** engine to run for 5 minutes to warm up.		N/A
Cue: 5 minutes has elapsed. Examiners Note: None Comments:				
5.1.13	*	**OPEN** 1CA1913 SBO COMPRESSOR DISCHARGE VALVE, to pressurize Control Air header.	**Locates and describes opening 1CA1913 SBO COMPRESSOR DISCHARGE VALVE.**	
Cue: 1CA1913, SBO COMPRESSOR DISCHARGE VALVE, is open. Examiners Note: None Comments:				
5.1.14	*	**OPEN** 1CA1886, BLACKOUT AIR COMPRESSOR ISOLATION.	**Locates and describes opening 1CA1886, BLACKOUT AIR COMPRESSOR ISOLATION.**	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Cue: 1CA1886, BLACKOUT AIR COMPRESSOR ISOLATION is open. Examiners Note: 1CA1886 is located near the entrance to Service Building on right side past the U2 mixing bottle. Comments:				
5.1.15	*	**OPEN** 2CA584, YARD CONTROL AIR SUPPLY VALVE, to pressurize Aux. Building 1A and 2A Control Air headers.	**Locates and describes opening 2CA584, YARD CONTROL AIR SUPPLY VALVE.**	
Cue: 2CA584, YARD CONTROL AIR SUPPLY VALVE is open. Examiners Note: 2CA584 is located near the entrance to Service Building on right side past the U2 mixing bottle. Comments:				
5.1.16	*	**PLACE** UNLOADER VALVE selector switch in RUN position.	**Returns to SBO Compressor Building and points out UNLOADER VALVE selector switch and describes placing it in RUN position.**	
Cue: Examiners Note: None. Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC IP-i **Revision No.:** 1

Task Title: Start the SBO Air Compressor IAW AB.LOOP-1

Task No.: N1150140501

Source: Bank X New ____ Mod ____

System: 056 Loss of Offsite Power

K/A Number / Description: AA 1.37 Ability to operate and/or monitor the following as they apply to Loss of Offsite Power: Instrument Air

K/A Rating RO 3.4 SRO N/A

Task Applicability: SRO Only ____ RO/SRO ____ AO/RO/SRO X Other ____

Time-Critical: Yes ____ No X **Alternate Path:** Yes ____ No X

Estimated Time to Complete: **15** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance X Actual Performance ____

Location: Classroom ____ Simulator ____ In-Plant X RCA ____

Required Materials: SC.OP-SO.CA-0001

Reference(s): SC.OP-SO.CA-0001

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

NITIAL CONDITIONS

- Both Salem Units have tripped due to a loss of off-site power. Equipment problems have resulted in NO Emergency Control Air Compressors (ECACs) running.

INITIATING CUE

- The Unit 2 CRS has directed you to start the Station Blackout Compressor (SBO) IAW SC.OP-SO.CA-0001, SBO Diesel Control Air Compressor, starting with Step 5.1.3.
- The Component Off Normal and Off Normal Tagged review has been performed SAT.
- It is **NOT** the summer operating period.
- The inspection covers have already been removed, Step 5.1.1 was performed SAT, and the inspection covers replaced IAW Step 5.1.2

PSEG Nuclear LLC

Job Performance Measure

START UP AND PARALLEL A ROD DRIVE M-G SET

JPM Number: 21-01 NRC IP-j

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**Shetrone/Militti/Westlake** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

JM 1. Task description and number, JPM description and number are identified.

JM 2. Knowledge and Abilities (K/A) references are included.

JM 3. Performance location specified. (in-plant, control room, or simulator)

JM 4. Initial setup conditions are identified.

JM 5. Initiating and terminating cues are properly identified.

JM 6. Task standards identified and verified by SME review.

JM 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

JM 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S1.OP-SO.RCS-0001 Rev. 36 Date Checked: 7/26/22

JM 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

N/A 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

JM 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

J. Militti	8/18/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an In-plant JPM.

SPECIAL INSTRUCTIONS

1. Marked up copies of S1.OP-SO.RCS-0001.
2. This JPM is located in the 84 ft. Unit 1 Switchgear Room (No RCA entry).

INITIAL CONDITIONS

- Salem Unit 1 is at 75% power.
- 12 Rod Drive M-G set is in service.
- 11 Rod Drive M-G Set tripped 3 days ago.
- Maintenance has completed repairs and is requesting 11 M-G set be placed in service.

INITIATING CUE

The CRS directs you to place 11 Rod Drive M-G set in service and parallel with 12 M-G set using section 4.4 of S1.OP-SO.RCS-0001, Rod Control System Operation.

TASK STANDARD:

The task is satisfactorily met when the applicant has started and paralleled 11 Rod Drive M-G set.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (optional):

[Page Left Intentionally Blank]

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue #1: Provide candidate MARKED UP copy of S1.OP-SO-RCS-0001. **Cue #2**: **READ TO APPLICANT:** No plant controls or equipment are to be operated during the performance of this JPM. All actions taken by the examinee should be clearly demonstrated and verbalized to the evaluator. **Examiners Note**: The candidate may wish to review all P&Ls prior to start. Comments: N/A				
4.4.1		**ENSURE** the following: 1. A AND B REACTOR TRIP BREAKERS are CLOSED. 2. Running Rod Drive Motor Generator Set GENERATOR LINE VOLTS is 260V (247V - 273V) AND GENERATOR LINE AMPS is ~80 DC AMPS.	Operator locates A AND B REACTOR TRIP BREAKERS and verifies the CLOSED indicator on each breaker. Operator locates 12 Rod Drive Motor Generator Set GENERATOR LINE VOLT AND GENERATOR LINE AMPS meters and checks readings.	
Cue #1: A and B Reactor Trip Breakers indicate CLOSED. **Cue #2**: 12 M-G set volts is reading 260V and Generator Line Amps is 80 DC amps. **Examiners Note**: 11 and 12 Rod Drive M-G sets are located in the 84 ft. Switchgear Room. Comments:				
4.4.2		**ALIGN** MG Set Controls for MG Set to be started IAW Attachment 1.	Operator aligns M-G set controls per Attachment 1.	
Cue: Cue the following for each component alignment when performed: ▪ Voltmeter selector switch is on 'A-B' position ▪ Voltage adjust is in 'Vertical' position ▪ Ammeter selector switch is in 'A' position ▪ Motor Breaker Control Switch target is 'GREEN' ▪ Motor Breaker is 'OPEN'. ▪ Synchronize switch is in 'OFF' position				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
▪ Generator Breaker Control Switch target is 'GREEN'. ▪ Generator Breaker is 'OPEN'. Examiners Note: None Comments:				
4.4.3	*	**CLOSE** MOTOR Breaker AND **ALLOW** at least 20 seconds for MG Set to reach rated speed.	Operator simulates closing 11 M-G set breaker by rotating the MOTOR Breaker switch to CLOSE.	
Cue: 11 M-G set MOTOR Breaker is CLOSED. Examiners Note: None Comments:				
4.4.4	*	**PRESS AND HOLD** GEN FIELD FLASH pushbutton **UNTIL** Generator voltage stops rising (GENERATOR LINE VOLTS should indicate ~260V).	Operator simulates pressing and holding the GEN FIELD FLASH pushbutton until Generator voltage stops rising.	
Cue: Generator Lines Volts indicates 260 Volts Examiners Note: None Comments:				
4.4.5		**CHECK** VOLTAGE ADJUST range of adjustment as follows:		
Cue: None Examiners Note: None Comments:				
4.4.5.1		**ADJUST** VOLTAGE ADJUST full counterclockwise to LOW.	Operator simulates adjusting VOLTAGE ADJUST full counterclockwise to LOW	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		(GENERATOR LINE VOLTS meter should indicate ~220V.)		
Cue: Generator Line Volts is 220 volts Examiners Note: None Comments:				
4.4.5.2		**ADJUST** VOLTAGE ADJUST full clockwise to HIGH. (GENERATOR LINE VOLTS meter should indicate ~300V)	Operator simulates adjusting VOLTAGE ADJUST full clockwise to HIGH.	
Cue: Generator Line Volts is 295 volts Examiners Note: None Comments:				
4.4.5.3		**ADJUST** VOLTAGE ADJUST **UNTIL** GENERATOR LINE VOLTS meter indicates ~260V.	Operator simulates adjusting VOLTAGE ADJUST (counterclockwise) until GENERATOR LINE VOLTS meter indicates ~260V.	
Cue: Generator Line Volts is 260 volts. **Examiners Note**: Adjusting VOLTAGE ADJUST in the COUNTERCLOCKWISE direction will LOWER the voltage. Comments:				
		NOTE DMM (OR equivalent, set to voltage) voltage readings will be less (by ≈ ½) than installed voltmeter readings.		
4.4.5.4		Direct Maintenance to obtain voltage reading on Running MG Set using a DMM (OR equivalent, set to voltage), at back of installed voltmeter.	Operator directs Maintenance to obtain voltage reading on Running MG Set using a DMM at back of installed voltmeter.	
Cue: Maintenance reports voltage reading on Running MG Set is 130 V. Examiners Note: None				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
4.4.5.5		Direct Maintenance to monitor voltage reading on In-coming MG Set using a DMM (**OR** equivalent, set to voltage), at back of installed voltmeter.	Operator directs Maintenance to monitor voltage reading on In-coming MG Set using a DMM at back of installed voltmeter.	
Cue: Maintenance is monitoring voltage reading on In-coming M-G set and reports value of 129.5V. Examiners Note: None Comments:				
4.4.5.6	*	**ADJUST** the VOLTAGE ADJUST **UNTIL** indicated voltage on DMM (**OR** equivalent, set to voltage), is 0-2.5 volts higher than the reading obtained in Step 4.4.5.4.	Operator simulates adjusting the VOLTAGE ADJUST in the CLOCKWISE direction until indicated voltage on DMM is 0-2.5 volts higher than the reading obtained in step 4.4.5.4.	
Cue: Maintenance reports voltage reading on In-coming MG Set is now **130.25 V** **Examiners Note**: Voltage from step 4.4.5.4 was **130 V**. Range is 0-2.5V higher than 130 V. I**F OPERATOR DOES NOT ADJUST, THEN DIRECT OPERATOR TO ADJUST VOLTAGE ABOVE 130 V (0-2.5V). Adjusting the VOLTAGE ADJUST switch in the CLOCKWISE direction will raise voltage.** Comments:				
4.4.5.7		**DIRECT** Maintenance to REMOVE DMM (**OR** equivalent, set to voltage), AND **CLOSE** panel doors.	Operator directs Maintenance to remove DMM AND close panel doors.	
Cue: Maintenance has removed DMM AND closed panel doors. Examiners Note: None. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		NOTE A single handle is shared between both MG set SYNCHRONIZE switches. The handle can only be removed when the switch is in the OFF position. The Generator Pistol Grip indication will remain GREEN when the Generator Breaker auto closes. This indicates a neutral position and is expected since the Pistol Grip has not been physically turned to the CLOSED position.		
		CAUTION Do **NOT** attempt to manually close generator output breaker. Manually closing generator output breaker could result in both MG Sets tripping. If the MG Sets are **NOT** closely synchronized, automatic breaker closure could take as long as 60 seconds		
4.4.6	*	**PLACE** MG Set SYNCHRONIZE Switch to ON, **AND OBSERVE** following: • GENERATOR Breaker automatically closes • Load is divided equally between MG Sets (~40 DC Amps each)	IF required, operator simulates moving the Synchronize Switch from 12 MG set and inserting into 11 MG set and places it in ON.	
Cue: 11 MG set generator breaker is closed and load stabilizes at 40 amps on each generator. Examiners Note: None. Comments:				
4.4.7		**PLACE** MG Set SYNCHRONIZE Switch to OFF.	Operator simulates rotating the SYNCHRONIZE Switch to the OFF position.	
Cue: SYNCHRONIZE Switch is in the OFF position. Examiners Note: None. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Terminating Cue		JPM COMPLETE		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC IP-j **Revision No.:** 0

Task Title: Start Up and Parallel a Rod Drive M-G Set

Task No.: N0015010104 **Source:**

System: 012 Reactor Protection System (SF7) Bank X New _____ Mod _____

K/A Number / Description: A4.07 Ability to manually operate and/or monitor in the control room: M/G set breakers

K/A Rating RO 3.8 SRO N/A

Task Applicability: SRO Only _____ RO/SRO _____ AO/RO/SRO X Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **15** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance X Actual Performance _____

Location: Classroom _____ Simulator _____ In-Plant X RCA _____

Required Materials: S1.OP-SO.RCS-0001

Reference(s): S1.OP-SO.RCS-0001

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Salem Unit 1 is at 75% power.
- 12 Rod Drive M-G set is in service.
- 11 Rod Drive M-G Set tripped 3 days ago.
- Maintenance has completed repairs and is requesting 11 M-G set be placed in service.

INITIATING CUE

The CRS directs you to place 11 Rod Drive M-G set in service and parallel with 12 M-G set using Section 4.4 of S1.OP-SO.RCS-0001, Rod Control System Operation.

PSEG Nuclear LLC

Job Performance Measure

PERFORM A RADIOACTIVE WASTE GAS RELEASE

JPM Number: 21-01 NRC IP-k

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**Shetrone/Militti/Westlake** **SME or Instructor**	**Date:**	8/18/22
Reviewed By:	**Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-SO.WG-0008 Rev. 27 Date Checked: 7/21/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

KH 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

K. Hantho	10/14/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. None, this is an In-plant JPM.

SPECIAL INSTRUCTIONS

1. Marked up copies of S2.OP-SO.WG-0008.
2. SPDS screenshot of MET DATA
3. This JPM is located at the 64 ft. Elev. Unit 2 Auxiliary Building (**RCA entry required**).

INITIAL CONDITIONS

- Both Salem Units are operating at 100% power
- 21 Waste Gas Decay Tank is in Standby at 90 psig
- 22 Waste Gas Decay Tank is in Holdup at 15 psig
- 23 Waste Gas Decay Tank is O/S at 15 psig
- 24 Waste Gas Decay Tank is O/S at 15 psig
- 2R41 Plant Vent radiation monitor is OPERABLE
- Plant vent flow rate and sample flow rate monitors are OPERABLE

INITIATING CUE

The CRS directs you to RELEASE 21 Waste Gas Decay Tank starting at Step 5.2.11 of S2.OP-SO.WG-0008, 21 Gas Decay Tank to Plant Vent.

- Sections 5.1 and Steps 5.2.1 through 5.2.10 are complete.

TASK STANDARD:

The task is satisfactorily met when the applicant has aligned the release path from 21 Waste Gas Decay Tank (WGDT) and released the radioactive gas from 21 WGDT, and upon high radiation alarm, terminates the release by locally closing the 2WG41.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue #1: Provide candidate MARKED UP copy of S2.OP-SO.WG-0008. **Cue #2**: **READ TO APPLICANT:** No plant controls or equipment are to be operated during the performance of this JPM. All actions taken by the examinee should be clearly demonstrated and verbalized to the evaluator. **Examiners Note**: The candidate may wish to review all P&Ls prior to start. Comments: N/A				
5.2.11		**PERFORM** the following preparations at Panel 104-2 for the GDT release:		
Cue: None. **Examiners Note**: 104-2 Panel is located at the 64 ft. Elev. inside the Auxiliary Building (RCA entry required). Comments:				
5.2.11.A		**ENSURE** 21 GDT is NOT in service		
Cue: 21 GDT is NOT in service. Examiners Note: None Comments:				
5.2.11.B		**ENSURE** 21 GDT is NOT selected for "Standby".		
Cue: 21 GDT is NOT selected for STANDBY. Examiners Note: None Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.2.11.C		**RECORD** 21 GDT "Initial Pressure" (PIS1036) on Attachment 3.		
Cue: 21 GDT pressure **(**PIS1036) indicates **90 psig**. Examiners Note: None Comments:				
5.2.11.D		**LATCH** AND **SET** 2WG41, GDT VENT CONT VALVE, as follows:		
Cue: None Examiners Note: None Comments:				
5.2.11.D.1	*	**TURN** 2WG41 Waste Gas Vent Valve Flow Bias fully counterclockwise until indicator <0%.	Operator simulates turning the Waste Gas Vent Valve Flow Bias knob fully COUNTER-CLOCKWISE until indicator is less than 0%.	
Cue: 2WG41 flow bias is less than 0% Examiners Note: None Comments:				
5.2.11.D.2	*	**POSITION** Selector Switch to OPEN AND **RELEASE** to AUTO position (spring return to AUTO).	Operator simulates positioning the selector switch to the OPEN position and then RELEASE to AUTO position.	
Cue: Selector switch move to the OPEN position and now is released back to the AUTO position. Examiners Note: 2WG41-SWT is the selector switch "2WG41 local control switch" at 104-2 panel Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.2.11.D.3	*	**TURN** 2WG41 Waste Gas Vent Valve Flow Bias clockwise until indicator ≥100%.	Operator simulates turning 2WG41 flow bias CLOCKWISE until indicator reads ≥ 100%	
Cue: 2WG41 flow bias indicates 100%. Examiners Note: None. Comments:				
N/A		**NOTE** 2WG41 will not immediately open.		N/A
Cue: None **Examiners Note**: The 2WG41 will NOT go full open until the flow bias is adjusted to 100%. Comments:				
5.2.11.D.4		**ENSURE** with Unit 2 Control Room that 2WG41 valve has lost the CLOSED indication AND audible alarm received	Operator calls Control Room to verify 2WG41 has lost the closed indication and audible alarm received.	
Cue: Control Room reports they've lost 2WG41 CLOSED indication and audible alarm was received. Examiners Note: None Comments:				
5.2.11.D.5	*	**TURN** 2WG41 Waste Gas Vent Valve Flow Bias fully counterclockwise until indicator <0%.	Operator simulates turning 2WG41 flow bias COUNTER-CLOCKWISE until indicator reads < 0%.	
Cue: 2WG41 flow bias reads less than 0%. Examiners Note: None Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.2.11.D.6	*	**ENSURE** with Unit 2 Control Room that 2WG41 has closed indication	Operator calls Control Room to verify 2WG41 has closed indication.	
Cue: Control Room reports that 2WG41 has CLOSED indication Examiners Note: None. Comments:				
5.2.11.D.7	*	**POSITION** 2WG41 Selector Switch to CLOSE.	Operator simulates positioning 2WG41 selector switch to CLOSE.	
Cue: 2WG41 selector switch is in the CLOSED position. Examiners Note: None. Comments:				
5.2.11.E		**PERFORM** an Independent Verification of the 2WG41 Waste Gas Vent Valve Flow Bias position, AND **RECORD** the Independent Verification on Attachment 1, Section 2.0.	Operator request another operator perform IV of the 2WG41 Waste Gas Vent Valve Flow Bias position.	
Cue: IV completed SAT. Examiners Note: None. Comments:				
5.2.11.F	*	**OPEN** 21WG31, GDT INLET VALVE	Operator simulates rotating valve COUNTER-CLOCKWISE to open 21WG31.	
Cue: 21WG31 is Open and Tag is released. Examiners Note: None. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		NOTE While 2WG41 is opening, pressure downstream of 2WG38 will fluctuate between "0" psig and "10" psig until 2WG41 is fully open (100%). Once 2WG41 is fully open, 2WG38 will throttle to maintain <8 psig during the release.		
5.2.11.G	*	Slowly **OPEN** 21WG34	Operator simulates slowly rotating COUNTER-CLOCKWISE to open 21WG34	
Cue: 21WG34 is Open. Examiners Note: None. Comments:				
5.2.11.H		**RECORD** Independent Verification (IV) of 21 GDT discharge valves 21WG31 and 21WG34 on Attachment 1, Section 3.0, prior to commencing 21 GDT release	Operator request IV of 21 GDT discharge valve 21WG31 and 21WG34.	
Cue: IVs completed SAT. Examiners Note: None. Comments:				
5.2.12		**COMMENCE** 21 GDT release as follows:		
Cue: None Examiners Note: None. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.2.12.A	*	**POSITION** 2WG41 Selector Switch to OPEN AND **RELEASE** to AUTO position (spring return to AUTO).	Operator simulates rotating the selector switch to the Open position and then release to the Auto position.	
Cue: Selector switch moved to the OPEN position and now is released back to the AUTO position. Examiners Note: None. Comments:				
5.2.12.B	*	SLOWLY **SET** 2WG41 Waste Gas Vent Valve Flow Bias to ≤100% position which corresponds to a maximum release rate of 32 SCFM	Operator simulates adjusting the 2WG41 Waste Gas Vent Valve Flow Bias to ≤100%.	
Cue: 2WG41 flow bias is 100%. Examiners Note: None. Comments:				
5.2.12.C		**PERFORM** an Independent Verification of the positioning of 2WG41 Waste Gas Vent Valve Flow Bias on Attachment 1, Section 4.0.	Operator request IV on 2WG41 flow bias position.	
Cue: IV completed SAT. Examiners Note: None. Comments:				
5.2.12.D		**RECORD** In Progress Release Data on Attachment 2, Section 5.1	Operator records Data on Attachment 2	
Cue: None. Examiners Note: None. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.2.13		**PERFORM** the following during 21 GDT release		
Cue: None. Examiners Note: None. Comments:				
5.2.13.A		**CALCULATE** 21 GDT Average Release Rate every 10 minutes on Attachment 3, AND **ADJUST** 2WG41 Waste Gas Vent Valve Flow Bias position as required based on results.	Operator reads step and action required every 10 minutes. No action is required at this time.	
Cue: None. **Examiners Note**: No action is required at this time, every 10 minutes. Comments:				
5.2.13.B		**RECORD** Meteorological Data in Attachment 2, Section 5.2. IF Meteorological Monitor is NOT OPERABLE, THEN **NOTIFY** the SM/CRS (UFSAR 7.7.2.12).	Applicant records the MET Data from the SPDS screenshot	
Cue: **PROVIDE** the following MET DATA screenshot to applicant Examiners Note: None. Comments:				
5.2.13.C		IF Plant Vent Flow Rate Monitor is inoperable, THEN **RECORD** Plant Vent Flow Rate Discharge Estimation on Attachment 4 at least once every four hours during GDT release	Applicant records step as N/A since Plant Vent Flow Rate Monitor is Operable.	
Cue: Plant Vent Flow Rate Monitor is OPERABLE. Examiners Note: None.				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
5.2.13.D		IF at any time during the release pressure downstream of 2WG38 is >8.0 psig (2PL8678), OR 2WG41 CLOSES, THEN **TERMINATE** the GDT release as follows: 1. **TURN** 2WG41 Waste Gas Vent Valve Flow Bias fully counter-clockwise until indicator is <0%. 2. **PLACE** 2WG41-SWT in CLOSE position, AND **ENSURE** 2WG41 is CLOSED. 3. **CLOSE** 22WG34. 4. **RECORD** Final Release Data in Attachment 2, Section 5.3. 5. **INITIATE** an NOTF to correct the malfunction.	Applicant reads step and continues.	
Cue: None. Examiners Note: None. Comments:				
5.2.14		When 21 GDT pressure ≈10 psig OR as directed by SM/CRS, **SECURE** the GDT release:		
Cue: 21 GDT pressure is now **10 psig**. Examiners Note: None. Comments:				
5.2.14.A	*	**TURN** 2WG41 Waste Gas Vent Valve Flow Bias fully counter-clockwise until indicator is <0%.	Operator simulates adjusting the Flow Bias fully COUNTER-CLOCKWISE until indicator is <0%.	
Cue: 2WG41 flow bias is now 0%.				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Examiners Note: None. Comments:				
5.2.14.B	*	**PLACE** 2WG41-SWT in CLOSE position, AND **ENSURE** 2WG41 is CLOSED	Operator simulates placing switch 2WG41-SWT in the CLOSED position and verifies that 2WG41 is CLOSED.	
Cue: Switch 2WG41-SWT is in the CLOSED position and the 2WG41 is now CLOSED. Examiners Note: None. Comments:				
5.2.14.C	*	**CLOSE** 21WG34	Operator simulates closing the 21WG34 by turning the valve in the CLOCKWISE direction.	
Cue: 21WG34 is now CLOSED. Examiners Note: None. Comments:				
5.2.14.D		**RECORD** Final Release Data in Attachment 2, Section 5.3		
Cue: None. Examiners Note: None. Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO X

Facility: Salem **JPM No.:** 21-01 NRC IP-k **Revision No.:** 0

Task Title: Perform a Radioactive Waste Gas Release

Task No.: N0715070104 **Source:**

System: 071 Waste Gas Disposal System (SF9) Bank X New _____ Mod _____

K/A Number / Description: A1.03 Ability to predict and/or monitor changes in parameters associated with operation of the Waste Gas Disposal System, including: Holdup tank pressure and level

K/A Rating RO 2.6 SRO N/A

Task Applicability: SRO Only _____ RO/SRO _____ AO/RO/SRO X Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: **20** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance X Actual Performance _____

Location: Classroom _____ Simulator _____ In-Plant X RCA X

Required Materials: S2.OP-SO.WG-0008

Reference(s): S2.OP-SO.WG-0008

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Both Salem Units are operating at 100% power
- 21 Waste Gas Decay Tank is in Standby at 90 psig
- 22 Waste Gas Decay Tank is in Holdup at 15 psig
- 23 Waste Gas Decay Tank is O/S at 15 psig
- 24 Waste Gas Decay Tank is O/S at 15 psig
- 2R41 Plant Vent radiation monitor is OPERABLE
- Plant vent flow rate and sample flow rate monitors are OPERABLE

INITIATING CUE

The CRS directs you to RELEASE 21 Waste Gas Decay Tank starting at Step 5.2.11 of S2.OP-SO.WG-0008, 21 Gas Decay Tank to Plant Vent.

- Sections 5.1 and Steps 5.2.1 through 5.2.10 are complete.

PSEG Nuclear LLC

Job Performance Measure

Perform Control Rod System Surveillance Test

JPM Number: 21-01 NRC Sim-a

Revision Number: 1

Date: 07/25/2022

Developed By:	**K. Hantho**	**Date:**	7/25/22
	Instructor		
Validated By:	**Westlake / Militti**	**Date:**	8/19/22
	SME or Instructor		
Reviewed By:	**M. Winkelspecht**	**Date:**	11/29/22
	Operations Representative		
Approved By:	**M. Wadusky**	**Date:**	11/29/22
	Training Department (Print/Sign)		

REVISION RECORD (Summary)

Revision Number	Date	Reason
01	7/25/22	Reformat JPM, update procedure rev and task standard after NUREG revision.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC	1. Task description and number, JPM description and number are identified.
RC	2. Knowledge and Abilities (K/A) references are included.
RC	3. Performance location specified. (in-plant, control room, or simulator)
RC	4. Initial setup conditions are identified.
RC	5. Initiating and terminating cues are properly identified.
TC	6. Task standards identified and verified by SME review.
RC	7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)
RC	8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:
RC	Procedure(s) S2.OP-ST.RCS-0001 Rev. 26 Date Checked: 7/25/22
RC	9. Pilot test the JPM:
	a. Verify cues both verbal and visual are free of conflict, and
	b. Ensure performance time is accurate.
NA	10. If the JPM cannot be performed as written with proper responses, then revise the JPM.
NA	11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11-28-22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-250 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is at 100% power.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
01	RD0045, Uncontrolled Rod Insert in Auto	N/A	N/A	N/A	ET-1	TRUE

1. Verify Overrides / Remotes:

ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition / Severity
01	B433 5 DI, Rod Bank Selector Sw in Auto				ET-1	ON

2. Verify Event Triggers:

ET#	Description	Command
1	KB433W1D, Rod bank Selector SW to Auto	

SPECIAL INSTRUCTIONS

1. Rod bank selector switch selected to CBC position.

2. Ensure ARPI screen up on Plant computer.

3. For efficiency, provide procedures to the operators up front to allow time to read and review.

4. If procedures not supplied ahead of time then ensure marked up surveillance and OP-AP-300-1001 procedure are available for students.

INITIAL CONDITIONS

- Unit 2 is at 100% power BOL.
- No major equipment is out of service and no Tech Specs are active.
- The rod control system surveillance is in progress.
- All sections are complete, except for exercising Control Bank D.

INITIATING CUE

- You are the Reactor Operator.
- The CRS directs you to complete the rod control system surveillance IAW S2.OP-ST.RCS-0001, Reactivity Control System Rod Control Assemblies.
- A Maintenance Technician is stationed at the Rod Control Power Cabinets (Relay Room)
- CRS directs that 15 steps of rod insertion will be performed to ensure each rod moves at least 10 steps.
- Notify the CRS when the testing is complete.
- Your evaluator will take care of all alarms not related to your task.

TASK STANDARD:

The task is satisfactorily met when the applicant has exercised Control Bank D at least 10 steps, and upon completion of the test, recognizes unexpected continuous rod movement and manually trips the reactor.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		RECORD the JPM Start Time when the operator acknowledges READY TO START JPM.	Operator reads P&Ls and reviews OP-AP-300-1001 prior to start	
Cue: Provide candidate MARKED UP surveillance procedure (S2.OP-ST.RCS-0001) and OP-AP-300-1001. Examiners Note: Comments:				
4.1.10.1		IF the reactor is critical, THEN **ENSURE** TAVG is within ±1°F of TREF.	Operator checks Tave/Tref recorder on 2RP4 and determines Tavg is within +/-1 F of Tref.	
Cue: Examiners Note: Comments:				
4.1.10.2	*	**PLACE** Bank Selector Switch in the "CBD" position.	Operator rotates selector switch clockwise to the CBD position	
Cue: Examiners Note: Comments:				
4.1.10.3		**ENSURE** GRP. SELECT "B" lights are illuminated on Rod Control System Power Cabinets 21BD and 22BD.	Contacts Maint Technician at Power Cabinets**.**	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Cue: Technician reports GRP SELECT 'B' lights are illuminated on Cabinets 21 BD and 22BD. Examiners Note: Comments:				
4.1.10.4		Insert Control Bank D 1 step.	Operator inserts one step.	
Cue: **Examiners Note**: Steps 4.1.10.4 and 4.1.10.5 will be performed three times consecutively prior to the 15 step insertion in step 6. Comments:				
4.1.10.5		Withdraw Control Bank D 1 step.	Operator withdraws one step.	
Cue: Examiners Note: 1 ☐ 2 ☐ 3 ☐ Comments:				
4.1.10.6	*	MANEUVER Control Bank D at least 10 steps in any one direction.	Operator inserts Control Bank D 15 steps.	
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
4.1.10.7		**ENSURE** each rod in Control Bank **D** indicated rod movement of at least 10 steps.	Operator monitors rod position on plant computer and determines D bank rods all moved at least 10 steps.	
Cue: Reactor Engineer reports rods indicate 10 step movement.(optional) **Examiners Note**: Operator may request to insert rods additional steps, IF so, just restate the request. Comments:				
4.1.10.8	*	RECORD CONTROL BANK D "Test Results" by initialing the SAT or UNSAT column using the Acceptance Criteria in Attachment 1, Rod Control Assembly Data.	Operator records test results as SAT.	
Cue: Examiners Note: Comments:				
4.1.10.9	*	RESTORE Control Bank D to the pre-test position.	Operator withdraws Bank D to previous position (ARO)	
Cue: Examiners Note: Comments:				
4.1.11.1		IF the reactor is critical, THEN ENSURE TAVG is within ±1°F of TREF.	Operator checks Tave/Tref recorder on 2RP4 and determines Tavg is within 1 F of Tref**.**	
Cue: Examiners Note:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
4.1.11.2		IF Turbine Power is ≤15%, THEN **PLACE** Bank Selector Switch in the "MAN" position.	N/A	
Cue:				
Examiners Note:				
Comments:				
ALTERNATE PATH STARTS HERE				
4.1.11.3	*	IF Turbine Power is >15%, THEN **PLACE** Bank Selector Switch in the "AUTO" OR "MAN" position as directed by the SM/CRS.	Operator determines that Rx power is > 15% and rotates selector switch counterclockwise to the AUTO position.	
Cue #1: IF asked, CRS directs rod bank selected to AUTO. **Cue #2**: After rod motion commences: IF operator recommends to CRS to place rods in Manual, THEN state; understand placing rods to manual. **Simulator Operator**: ENSURE **ET-1** is TRUE when rods selected to Auto. This will insert MALF: RD0045, Uncontrolled Rod Insertion in AUTO AND MANUAL. **Examiners Note**: The operator may refer to S2.OP-AB.ROD-0003, Continuous Rod Motion and take the actions in the AB to manually trip the reactor. **JPM Complete once Reactor is Tripped.** Comments:				
	*	Operator announces that rods are stepping in and no runback in progress. Operator places rod bank switch to Manual and reports rod motion has NOT stopped. Operator recognizes rod insertion is continuing	Operator Manually trips the Reactor.	
Cue:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Examiners Note: Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-a **Revision No.:** 1

Task Title: Perform Rod Control System Surveillance Test

Task No.: N1140340401

Source: Bank X New _____ Mod _____

System: 001 Control Rod Drive System (SF1)

K/A Number / Description: A4.16 Ability to manually operate and/or monitor in the control room: Rod speed and direction

K/A Rating RO 3.8 SRO N/A

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **15** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: S2.OP-ST.RCS-0001

Reference(s): S2.OP-ST.RCS-0001 OP-AP-300-1001

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Unit 2 is at 100% power BOL.
- No major equipment is out of service and no Tech Specs are active.
- The rod control system surveillance is in progress.
- All sections are complete, except for exercising Control Bank D.

INITIATING CUE

- You are the Reactor Operator.
- The CRS directs you to complete the rod control system surveillance IAW S2.OP-ST.RCS-0001, Reactivity Control System Rod Control Assemblies.
- A Maintenance Technician is stationed at the Rod Control Power Cabinets (Relay Room)
- CRS directs that 15 steps of rod insertion will be performed to ensure each rod moves at least 10 steps.
- Notify the CRS when the testing is complete.
- Your evaluator will take care of all alarms not related to your task.

PSEG Nuclear LLC

Job Performance Measure

Transfer To Hot Leg Recirculation

JPM Number: 21-01 NRC Sim-b

Revision Number: 03

Date: 7/26/222

Developed By:	**K. Hantho**	**Date:**	7/26/22
	Instructor		
Validated By:	**Westlake/Militti**	**Date:**	8/19/22
	SME or Instructor		
Reviewed By:	**M. Winkelspecht**	**Date:**	11/29/22
	Operations Representative		
Approved By:	**M. Wadusky**	**Date:**	11/29/22
	Training Department (Print/Sign)		

REVISION RECORD (Summary)

Revision Number	Date	Reason
03	7/26/22	Update and validate to new JPM format.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

1. Task description and number, JPM description and number are identified.
2. Knowledge and Abilities (K/A) references are included.
3. Performance location specified. (in-plant, control room, or simulator)
4. Initial setup conditions are identified.
5. Initiating and terminating cues are properly identified.
6. Task standards identified and verified by SME review.
7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)
8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

   Procedure(s) 2-EOP-LOCA-4 Rev. 41 Date Checked: 7/26/22

9. Pilot test the JPM:
   a. Verify cues both verbal and visual are free of conflict, and
   b. Ensure performance time is accurate.
10. If the JPM cannot be performed as written with proper responses, then revise the JPM.
11. When JPM is revalidated, SME or instructor sign and date JPM cover page

SME/Instructor Date

SME/Instructor Date

SME/Instructor Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-251 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is shutdown.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
01	RC0001A, RCS rupture of RC loop 21	N/A	N/A	N/A	N/A	TRUE
02	SJ0062A, 21 SI pump trip	N/A	N/A	N/A	N/A	TRUE
03	Rh0026a 21 RHR PUMP TRIP	N/A	N/A	N/A	N/A	TRUE
ID# OVRD or REMT	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity
01	RH26D, 21 RHR pump breaker control power					OFF
02	RH27D, 21 RHR pump rack out					TAGGED

ET#	Description	Command
none		

SPECIAL INSTRUCTIONS

1. Place Bezel cover on 21 RHR pump as C/T

INITIAL CONDITIONS

- You are the **incoming** crew.
- Unit 2 experienced a Large Break LOCA.
- The previous crew completed the actions to transfer to Cold Leg Recirculation IAW EOP-LOCA-3 and have transitioned back to EOP-LOCA-1.
- 21 SI pump tripped following completion of LOCA-3 and is being investigated.
- 21 RHR pump is C/T for scheduled maintenance prior to the event.

INITIATING CUE

- You are the **Reactor Operator**.
- The CRS has directed you to **IMPLEMENT** 2-EOP-LOCA-4, TRANSFER TO HOT LEG RECIRCULATION.
- Your evaluator will respond to all alarms not related to your task.

TASK STANDARD:

The task is satisfactorily met when the applicant has aligned the plant for Hot leg Recirculation.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

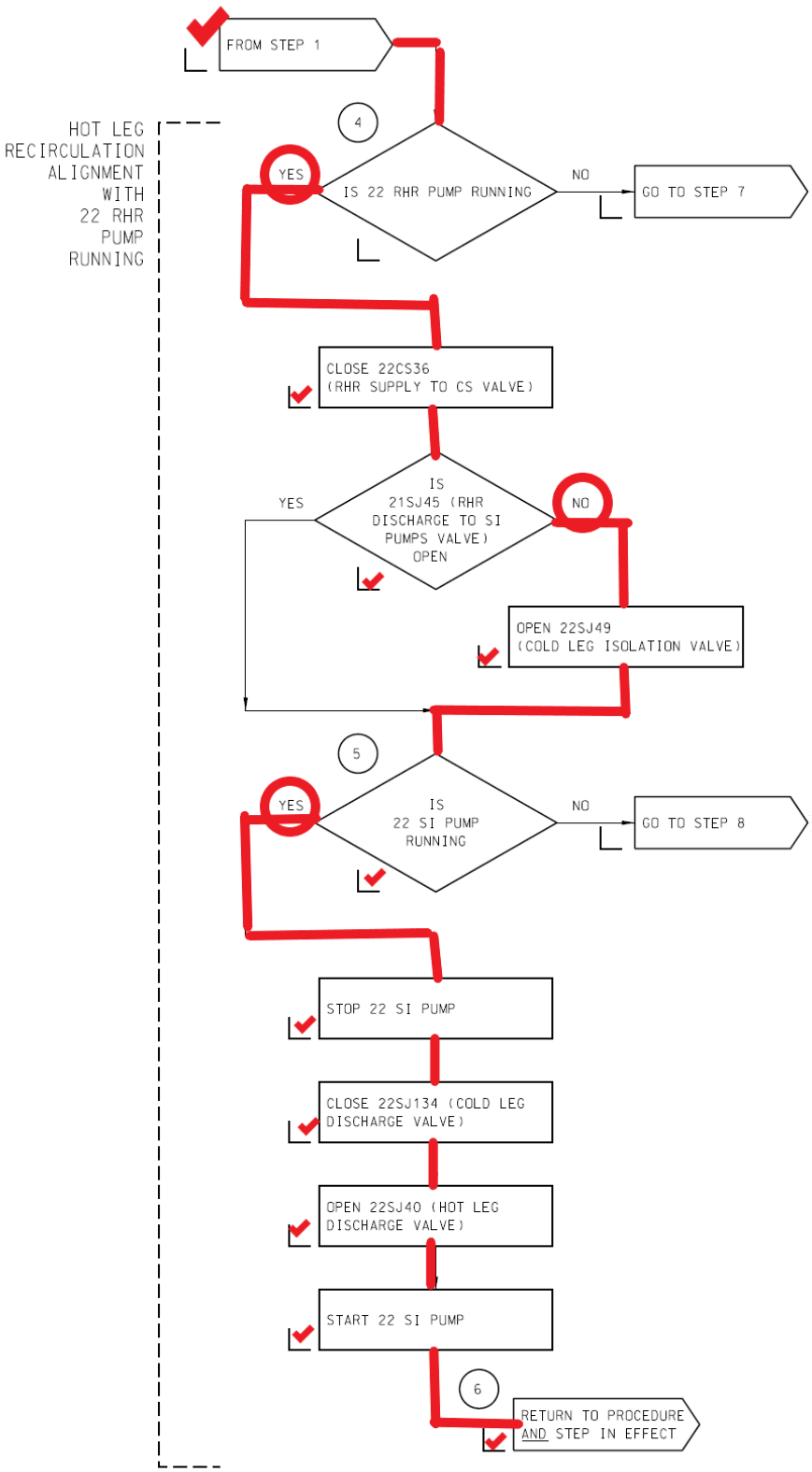
If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY:


HOT LEG RECIRCULATION ALIGNMENT WITH 22 RHR PUMP RUNNING
FROM STEP 1
4
IS 22 RHR PUMP RUNNING
YES
NO
GO TO STEP 7
CLOSE 22CS36 (RHR SUPPLY TO CS VALVE)
IS 21SJ45 (RHR DISCHARGE TO SI PUMPS VALVE) OPEN
YES
NO
OPEN 22SJ49 (COLD LEG ISOLATION VALVE)
5
IS 22 SI PUMP RUNNING
YES
NO
GO TO STEP 8
STOP 22 SI PUMP
CLOSE 22SJ134 (COLD LEG DISCHARGE VALVE)
OPEN 22SJ40 (HOT LEG DISCHARGE VALVE)
START 22 SI PUMP
6
RETURN TO PROCEDURE AND STEP IN EFFECT

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate sheet of 2-EOP-LOCA-4. Examiners Note: None Comments:				
LOCA-4 1.0		ARE **BOTH** RHR PUMPS RUNNING?	Determines that ONLY 22 RHR Pump is running. (21 RHR pump is C/T) **GOES TO Step 4.**	
Cue: Examiners Note: Comments:				
4.0		IS 22 RHR Pump Running	YES, determines that 22 RHR Pump is running.	
Cue: Examiners Note: Comments:				
4.0 Cont'd	*	CLOSE 22CS36 (RHR SUPPLY TO CS VALVE)	Operator depresses CLOSE PB for 22CS36 (RHR SUPPLY TO CS VALVE) and verifies CLOSE PB illuminates.	
Cue: Examiners Note:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
ALTERNATE PATH STARTS HERE				
4.0 Cont'd		IS 21SJ45 (RHR DISCHARGE TO SI PUMPS VALVE) OPEN	NO.	
Cue: Examiners Note: Comments:				
4.0 Cont'd	*	OPEN 22SJ49 (COLD LEG ISOLATION VALVE)	Operator depresses OPEN PB for 22SJ49 (COLD LEG ISOLATION VALVE) and verifies OPEN PB illuminates.	
Cue: Examiners Note: Comments:				
5.0		IS 22 SI Pump Running	Yes	
Cue: Examiners Note: Comments:				
5.0 Cont'd	*	STOP 22 SI Pump	Operator depresses 22 SI Pump STOP PB on 2CC1 and verifies STOP PB illuminates.	
Cue:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Examiners Note: Comments:				
5.0 Cont'd	*	CLOSE 22SJ134 (Cold Leg Discharge Valve)	Operator depresses CLOSE PB for 22SJ134 (Cold Leg Discharge Valve) on 2CC1 and verifies CLOSE PB illuminates.	
Cue: Examiners Note: Comments:				
5.0 Cont'd	*	OPEN 22SJ40 (Hot Leg Discharge Valve)	Operator Opens 22SJ40 (Hot Leg Discharge Valve) by inserting key and rotating; verifies OPEN PB illuminates.	
Cue: Examiners Note: Comments:				
5.0 Cont'd	*	START 22 SI Pump	Operator depresses START PB for 22 SI Pump and verifies START PB illuminates.	
Cue: Examiners Note: Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-b **Revision No.:** 3

Task Title: Transfer To Hot Leg Recirculation

Task No.: 1150110501

Source: Bank X New _____ Mod _____

System: 006 Emergency Core Cooling System (SF2)

K/A Number / Description: A4.05 Ability to manually operate and/or monitor in the control room: Transfer of ECCS flowpaths

K/A Rating RO 4.1 SRO N/A

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **6** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: 2-EOP-LOCA-4

Reference(s): 2-EOP-LOCA-4

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- You are the **incoming** crew.
- Unit 2 experienced a Large Break LOCA.
- The previous crew completed the actions to transfer to Cold Leg Recirculation IAW EOP-LOCA-3 and have transitioned back to EOP-LOCA-1.
- 22 SI pump tripped following completion of LOCA-3 and is being investigated.
- 21 RHR pump is C/T for scheduled maintenance prior to the event.

INITIATING CUE

- You are the **Reactor Operator**.
- The CRS has directed you to **IMPLEMENT** 2-EOP-LOCA-4, TRANSFER TO HOT LEG RECIRCULATION.
- Your evaluator will respond to all alarms not related to your task.

PSEG Nuclear LLC

Job Performance Measure

Depressurize RCS using Aux spray IAW EOP-LOCA-2

JPM Number: 21-01 NRC Sim-c

Revision Number: 00

Date: 10/12/22

Developed By:	**K. Hantho**	**Date:**	10/12/22
	Instructor		
Validated By:	**Gallagher/Saulsbery**	**Date:**	11/21/22
	SME or Instructor		
Reviewed By:	**M. Winkelspecht**	**Date:**	11/29/22
	Operations Representative		
Approved By:	**M. Wadusky**	**Date:**	11/29/22
	Training Department (Print/Sign)		

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	10/12/22	New JPM

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

EG 1. Task description and number, JPM description and number are identified.

EG 2. Knowledge and Abilities (K/A) references are included.

EG 3. Performance location specified. (in-plant, control room, or simulator)

EG 4. Initial setup conditions are identified.

EG 5. Initiating and terminating cues are properly identified.

EG 6. Task standards identified and verified by SME review.

EG 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

EG 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) 2-EOP-LOCA-2 Rev. 41 Date Checked: 11/21/22

EG 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

Ed Gallagher	11/21/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-252 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is shutdown.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
01	RC0002 RCS Leak in Containment					350
02	VL0016 22SJ54 Fails to Position					0
03	VL0016 21SJ54 Fails to Position					0
04	VL0016 23SJ54 Fails to Position					0
05	VL0016 24SJ54 Fails to Position					0
06	PR0019D PZR Spray valve 2PS3 Fails Closed					True
07	VL0298 2PR2 Fails to position					0.00

Place a Bezel Cover over 2PR1, to indicate isolated status

SPECIAL INSTRUCTIONS

1. Marked up copy of 2-EOP-LOCA-2 on center console, complete thru step 19

INITIAL CONDITIONS

- The reactor was tripped when a RCS leak occurred.
- All RCPs have been stopped
- 2PR1 is inoperable with 2PR6 isolated with power removed
- The operating crew has progressed through the EOP's and is now in 2-EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION

INITIATING CUE

- You are the Reactor Operator.
- The CRS directs you to depressurize the RCS to minimize subcooling IAW Step 20 of 2-EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION
- Notify the CRS when Step 20 is completed.
- Your evaluator will take care of all alarms not related to your task.

TASK STANDARD:

The task is satisfactorily met when the applicant has depressurized RCS using Auxiliary spray.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

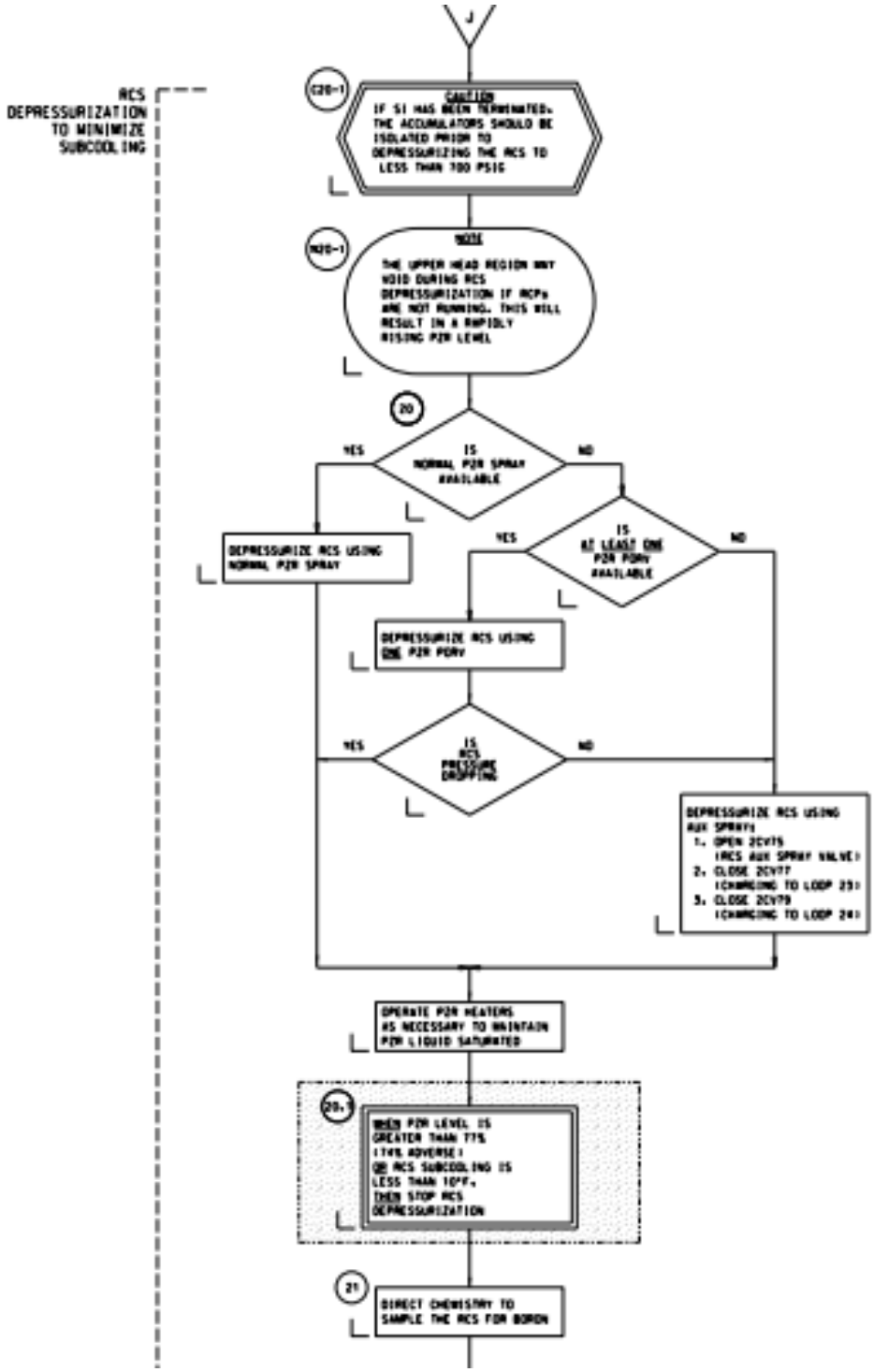
The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY (if required):

Flowchart portion below



RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate MARKED UP sheet of 2-EOP-LOCA-2. Examiners Note: Comments:				
C20-1		CAUTION IF SI HAS BEEN TERMINATED, THE ACCUMULATORS SHOULD BE ISOLATED PRIOR TO DEPRESSURIZING THE RCS TO LESS THAN 700 PSIG	Verifies 21-24SJ54s closed	
Cue: Examiners Note: The 21-24 SJ54s are closed Comments:				
N20-1		NOTE THE UPPER HEAD REGION MAY VOID DURING RCS DEPRESSURIZATION IF RCPs ARE NOT RUNNING THIS WILL RESULT IN A RAPIDLY RISING PZR LEVEL.	Acknowledge the Note	
Cue: Examiners Note: The 21-24 SJ54s are closed Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
20		IS NORMAL PZR SPRAY AVAILABLE	NO.	
Cue: Examiners Note: With no RCPs running, normal spray is not available Comments:				
20 CONT'D	*	IS AT LEAST ONE PZR PORV AVAILABLE	YES.	
Cue: Examiners Note: Comments:				
20 CONT'D	*	DEPRESSURIZE RCS USING ONE PZR PORV	Operator Opens the 2PR2 to attempt to depressurize the RCS but it will not open. 2PR1 remains isolated and unavailable.	
Cue: Examiners Note: Comments:				
ALTERNATE PATH STARTS HERE				
20 CONT'D	*	IS RCS PRESSURE DROPPING	NO.	
Cue: Examiners Note: Operator observes valve not repositioning. May attempt to depress the OPEN PB again. Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
20 CONT'D	*	DEPRESSURIZE RCS USING AUX SPRAY: 1. OPEN 2CV75 (RCS AUX SPRAY VALVE) 2. CLOSE 2CV77 (CHARGING TO LOOP 23) 3. CLOSE 2CV79 (CHARGING TO LOOP 24)	Operator opens 2CV75 and closes 2CV77 and 79. Operator monitors RCS pressure and verifies pressure is lowering.	
Cue : Examiners Note: Comments:				
20 CONT'D		OPERATE PZR HEATERS AS NECESSARY TO MAINTAIN PZR LIQUID SATURATED	Operator will check PZR temperature and determine if pressurizer heaters are required to be energized	
Cue : When the operator checks that RCS pressure is lowering, and determines whether or not heaters need to be energized then, provide the following cue: • RCS subcooling is now <10°F Examiners Note: Comments:				
20.1	*	WHEN PZR LEVEL IS GREATER THAN 77% (74% ADVERSE) OR RCS SUBCOOLING LESS THAN 10°F THEN STOP RCS DEPRESSURIZATION	Operator should realign aux spray to normal as follows: Operator closes 2CV75 and opens 2CV77 or 79.	
Cue: Examiners Note: Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-c **Revision No.:** 00

Task Title: Depressurize RCS using Aux spray IAW EOP-LOCA-2

Task No.: N1150090501 **Source:**

System: 010 PZR PCS Pressurizer Pressure Control System Bank _____ New X Mod _____

K/A Number / Description: A4.05 Ability to manually operate and/or monitor in the control room: PZR Auxiliary Spray Valves

K/A Rating RO 3.2 SRO N/A

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **5** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: 2-EOP-LOCA-2

Reference(s): 2-EOP-LOCA-2

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- The reactor was tripped when a RCS leak occurred.
- All RCPs have been stopped
- 2PR1 is inoperable with 2PR6 isolated with power removed
- The operating crew has progressed through the EOP's and is now in 2-EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION

INITIATING CUE

- You are the Reactor Operator.
- The CRS directs you to depressurize the RCS to minimize subcooling IAW Step 20 of 2-EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION
- Notify the CRS when Step 20 is completed.
- Your evaluator will take care of all alarms not related to your task.

PSEG Nuclear LLC

Job Performance Measure

INITIATE BLEED AND FEED USING REACTOR HEAD VENTS

JPM Number: 21-01 NRC Sim-d

Revision Number: 06

Date: 7/26/22

Developed By:	**K. Hantho**	**Date:**	7/26/22
	Instructor		
Validated By:	**Westlake / Militti**	**Date:**	8/19/22
	SME or Instructor		
Reviewed By:	**M. Winkelspecht**	**Date:**	11/29/22
	Operations Representative		
Approved By:	**M. Wadusky**	**Date:**	11/29/22
	Training Department (Print/Sign)		

REVISION RECORD (Summary)

Revision Number	Date	Reason
06	7/26/22	Update to new JPM format, update to NUREG 1021 changes.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) 2-EOP-FRHS-1 Rev. 41 Date Checked: 7/26/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11/28/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-253 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is shutdown.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
01	AF0181A, 21 AFW pump trip	N/A	N/A	N/A	N/A	TRUE
02	AF0181B, 22 AFW pump trip	N/A	N/A	N/A	N/A	TRUE
03	AF0183, 23 AFW pump overspeed trip	N/A	N/A	N/A	N/A	TRUE
04	VL0298, 2PR2 fails to position	N/A	N/A	N/A	N/A	0%

1. Verify the following OVERRIDES / REMOTES:

ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition / Severity
01	AF20D, 21 AFW control power	N/A	N/A	N/A	N/A	OFF
02	AF25D, 22 AFW control power	N/A	N/A	N/A	N/A	OFF

2. These malfunctions will simulate a total loss of all feed water and reaching the Bleed and Feed criteria of 3 SGs < 20% WR levels IAW FRHS-1. **ALTERNATE PATH**: 2PR2 will fail to open requiring the use of Rx Head Vent valves.

SPECIAL INSTRUCTIONS

1. **ENSURE** four (4) Reactor Head Vent keys are available for the evaluator.
2. Extra Instructor to respond to alarms.

INITIAL CONDITIONS

- Unit 2 was manually tripped at 90% power due to elevated main turbine vibrations IAW S2.OP-AB.TL-0001.
- An inadvertent feedwater isolation actuation occurred resulting in the loss of all main feedwater.
- Loss of all AFW flow occurred when all AFW pumps tripped.
- The crew has entered 2-EOP-FRHS-1 due to valid RED Path on Heat Sink CFST based on all SG NR levels < 9% and AFW flow < 22E4 lbm/hr.
- Plant conditions have continued to deteriorate and three (3) steam generator Wide Range levels are now less than 20%.
- MSPI AFW pump failed to start.
- The CRS is implementing the CAS action to Initiate Bleed and Feed.

INITIATING CUE

- You are the Reactor Operator.
- The CRS has directed you to Initiate Bleed and Feed starting at **Step 11** IAW 2-EOP-FRHS-1, Response to Loss of Secondary Heat Sink.
- Your evaluator will respond to all alarms not related to your task.

TASK STANDARD:

The task is satisfactorily met when the applicant has actuated Safety Injection, opened one Pressurizer PORV, and opened 2RC40 - 43 RCS head vent valves.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate MARKED UP sheet of 2-EOP-FRHS-1. Examiners Note: None Comments:				
C11-1		**CAUTION** Steps 11 thru 14 Must Be Performed Quickly in order to establish RCS Heat Removal by RCS Bleed and Feed.	Operator reads the Caution and continues on.	
11	*	Actuate Both trains of SI	Operator initiates SI on both Trains by inserting key and turning switch clockwise until the SI Operate bezel illuminates and the SI Reset bezel extinguishes.	
Cue: Examiners Note: Operator only needs to initiate SI on one Train. Comments:				
12		Is centrifugal Charging Pump flow indicated on SI systems Charging flow meter	YES. Operator reports BIT flow is established by indication of flow on Charging flowmeter. (During validations flow was about 270 gpm)	
Cue: Examiners Note: Comments:				
12 Cont'd		Are all SI pumps Running	YES. Operator reports that both SI pumps are running by START PBs illuminated Red.	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Cue: Examiners Note: Comments:				
13		Is power available to BOTH PZR PORV stop valves	Yes operator reports power is available to both Stop valves	
Cue: Examiners Note: Comments:				
13 Cont'd		Are ALL PZR PORV Stop Valves OPEN	Operator reports that both PZR PORV Stop Valves are OPEN as indicated by 2PR6 and 2PR7 OPEN bezels illuminated.	
Cue: Examiners Note: Comments:				
ALTERNATE PATH STARTS HERE				
13 Cont'd	*	Open ALL PZR PORVs	Operator places both PZR PORVs in Manual and depresses OPEN PBs for 2PR1 and 2PR2. Operator reports that 2PR2 will NOT OPEN.	
Cue: **Examiners Note:** The PZR PORVS may be cycling during this time. The operator should determine that both PORVs are required to be fully open in Manual and maintained open and not credit auto operation.				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
14		Are ALL PZR PORVs Open	NO. Operator reports that 2PR2 will NOT OPEN.	
Cue: Examiners Note: Comments:				
14 Cont'd	*	Open 2RC40 Thru 2RC43 (Reactor Head Vents)	At the 2RP2 backpanel, Operator obtains keys and inserts key into each keyswitch one at a time and turns clockwise until the OPEN indicator illuminates.	
Cue: **Examiners Note**: Lead Evaluator needs to provide the four (4) keys to the Operator. Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-d **Revision No.:** 6

Task Title: Initiate Bleed and Feed Using Reactor Head Vents

Task No.: 1150290501

Source: Bank X New _____ Mod _____

System: 002 Reactor Coolant System (SF4P)

K/A Number / Description: A2.04 Ability to (a) predict the impacts of the following on the Reactor Coolant System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of heat sinks

K/A Rating RO 4.5 SRO 4.5

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **8** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: 2-EOP-FRHS-1

Reference(s): 2-EOP-FRHS-1

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Unit 2 was manually tripped at 90% power due to elevated main turbine vibrations IAW S2.OP-AB.TL-0001.
- An inadvertent feedwater isolation actuation occurred resulting in the loss of all main feedwater.
- Loss of all AFW flow occurred when all AFW pumps tripped.
- The crew has entered 2-EOP-FRHS-1 due to valid RED Path on Heat Sink CFST based on all SG NR levels < 9% and AFW flow < 22E4 lbm/hr.
- Plant conditions have continued to deteriorate and three (3) steam generator Wide Range levels are now less than 20%.
- MSPI AFW pump failed to start.
- The CRS is implementing the CAS action to Initiate Bleed and Feed.

INITIATING CUE

- You are the Reactor Operator.
- The CRS has directed you to Initiate Bleed and Feed starting at **Step 11** IAW 2-EOP-FRHS-1, Response to Loss of Secondary Heat Sink.
- Your evaluator will respond to all alarms not related to your task.

PSEG Nuclear LLC

Job Performance Measure

Establish FW flow IAW FRHS 1 using Condensate pump

JPM Number: NRC SIM-e

Revision Number: 0

Date: 08/19/2022

Developed By:	**K. Hantho** **Instructor**	**Date:**	8/19/22
Validated By:	**Gallagher/Saulsbery** **SME or Instructor**	**Date:**	11/21/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	8/19/22	New JPM

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

EG 1. Task description and number, JPM description and number are identified.

EG 2. Knowledge and Abilities (K/A) references are included.

EG 3. Performance location specified. (in-plant, control room, or simulator)

EG 4. Initial setup conditions are identified.

EG 5. Initiating and terminating cues are properly identified.

EG 6. Task standards identified and verified by SME review.

EG 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

EG 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) 2-EOP-FRHS-1 rev 42 Date Checked: 11/21/22

EG 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

Ed Gallagher	11/21/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

1. IC-254

Malfunctions

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
BF0105A	21 STM Gen Feed Pump trip	N/A	N/A	N/A	N/A	TRUE
BF0105A	22 STM Gen Feed Pump trip	N/A	N/A	N/A	N/A	TRUE
AF0183 AF0181A &B	21-23 AFW pump trips	N/A	N/A	N/A	N/A	TRUE

Remotes

ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition/ Severity
BF01A through BF08A	21 through 24 BF19s and 40s	N/A	-1	30s	RT-1 through 8	100

SPECIAL INSTRUCTIONS

1. FRHS-1 EOP

INITIAL CONDITIONS

- Salem Unit 2 is in Mode 3.
- No SGFP's or AFW pumps are available.
- Actions are being implemented IAW 2-EOP-FRHS-1.
- The Crew has completed actions through step 7.1

INITIATING CUE

- CRS directs you to continue performing actions of FHRS-1 at Note 7-2

TASK STANDARD:

The task is satisfactorily met when the applicant has selected and depressurized a steam generator using the steam dumps, to established flow with a condensate pump and places steam dumps in auto to stabilize pressure.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

* **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be sued to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps I this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Note 7-2		21 AND 23 SGS SHOULD BE STEAMED LAST TO MAXIMIZE AVAILABILITY OF A STEAM SUPPLY FOR 23 AFW PP	Candidate should pick 22 or 24 to depressurize	
Cue: Examiners Note: <u>Comments:</u>				
Note 7-3		THE FOLLOWING STEPS MAY RESULT IN A LOW STEAMLINE DIFF PRESS SI	Acknowledge note	
Cue: Examiners Note: <u>Comments:</u>				
7.2		WHEN SELECTED SG PRESSURE IS LESS THAN 575 PSIG AND SELECTED SG WIDE RANGE LEVEL IS RISING. THEN STABILIZE SELECTED SG IS LESS THAN 575 PSIG.	Reads CAS	
Cue: Examiners Note: <u>Comments:</u>				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
7.2 Contd		SELECT ONLY ONE SG FOR DEPRESSURIZATION TO LESS THAN 575 PSIG	Chooses 22 or 24	
Cue: Examiners Note: Comments:				
7.3		ARE CONDENSER STEAM DUMPS AVAILABLE	Should answer yes path	
Cue: Examiners Note: Comments:				
7.3 contd	*	ACTUATE MAIN STEAM ISOLATION FOR ALL LOOPS EXCEPT SELECTED SG	Actuate MSLI for 21&23 and 22or 24 MSLines	
Cue: Examiners Note: Comments:				
7.3 contd	*	PLACE STEAM DUMPS IN MANUAL		
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
7.3 contd		ADJUST STEAM PRESSURE VALVE DEMAND TO 0%		
Cue: Examiners Note: Comments:				
7.3 contd	*	PLACE STEAM DUMPS IN MS PRESS CONTROL		
Cue: Examiners Note: Should be at 0 valve demand for MS pressure control Comments:				
7.4	*	WHEN RCS TAVG LOW LOW IS REACHED THEN DEPRESS BYPASS TAVG PUSHBUTTON	Bypasses if/when reqd	
Cue: Examiners Note: Comments:				
7.4 contd	*	ADJUST STEAM PRESSURE VALVE DEMAND TO ESTABLISH MAX RATE	Adjusts to at least 25%	
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
7.4 contd	*	SEND OPERATOR TO OPEN SELECTED BF40	Dispatches operator	
Cue: when asked tell candidates operator being dispatched Examiners Note: Insert RT for valve selected Comments:				
7.4 contd	*	OPEN SELECTED BF13	Opens BF13	
Cue: Examiners Note: Comments:				
7.4 contd		Is selected BF13 open	Will be yes but if they do not push button properly may dispatch operator if reqd.	
Cue: Examiners Note: Comments:				
7.4 contd		Release selected BF22	Verifies open	
Cue: Examiners Note: Comments:				

7.4 contd		Open 21 and 22 CN48	Opens valves	
Cue: Examiners Note: <u>Comments:</u>				
7.4 contd		Close 21&22CN32	Performs this task	
Cue: Examiners Note: <u>Comments:</u>				
7.5	*	IS CONDENSATE FLOW ESTABLISHED TO ANY SG	When **YES AND WR LEVEL IS RISING** then Steam dumps should be placed in auto	
Cue: Once Steam dumps in auto and WR level rising state JPM complete. Examiners Note: <u>Comments:</u>				

RECORD JPM Stop Time: _____

Form 3.2-3 Job Performance Measure

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** NRC SIM-e **Revision No.:** 0

Task Title: Establish FW flow IN FRHS using Condensate pump

Task No.: N1150290501

System: Radiation Control

K/A Number / Description: 041 A4.02 Ability to manually operate and/or monitor in the control room: Steam dump cooldown valves

K/A Rating RO 3.6 SRO 3.6

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes _____ No X

Estimated Time to Complete: 12 Minutes **Actual Time Used:** _____ Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____

Required Materials: EOP-FRHS-1

Reference(s): EOP-FRHS-1

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

Salem Unit 2 is in Mode 3.

No SGFP's or AFW pumps are available.

Actions are being implemented IAW 2-EOP-FRHS-1.

The Crew has completed actions through step 7.1

INITIATING CUE

CRS directs you to continue performing actions of FRHS-1 at Note 7.2

PSEG Nuclear LLC

Job Performance Measure

Perform 22 CFCU surveillance test IAW S2.OP-ST.CBV-0003

JPM Number: 21-01 NRC Sim-f

Revision Number: 1

Date: 07/26/2022

Developed By:	**K. Hantho**	**Date:**	7/26/22
	Instructor		
Validated By:	**Westlake/Militti**	**Date:**	8/19/22
	SME or Instructor		
Reviewed By:	**M. Winkelspecht**	**Date:**	11/29/22
	Operations Representative		
Approved By:	**M. Wadusky**	**Date:**	11/29/22
	Training Department (Print/Sign)		

REVISION RECORD (Summary)

Revision Number	Date	Reason
01	07/26/22	Reformat JPM, update procedure rev and task standard after NUREG revision and validation performed comments incorporated.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

1. Task description and number, JPM description and number are identified.
2. Knowledge and Abilities (K/A) references are included.
3. Performance location specified. (in-plant, control room, or simulator)
4. Initial setup conditions are identified.
5. Initiating and terminating cues are properly identified.
6. Task standards identified and verified by SME review.
7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)
8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

   Procedure(s) S2.OP-ST.CBV-0003 Rev. 20 Date Checked: 10/12/22
9. Pilot test the JPM:
   a. Verify cues both verbal and visual are free of conflict, and
   b. Ensure performance time is accurate.
10. If the JPM cannot be performed as written with proper responses, then revise the JPM.
11. When JPM is revalidated, SME or instructor sign and date JPM cover page

SME/Instructor Date

SME/Instructor Date

SME/Instructor Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-255 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is at 100% power.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

1. Verify Malfunctions:

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
01	VL0570, 22SW223 fails to position	00:00:15	N/A	N/A	ET-1	89

2. Verify Overrides / Remotes:

ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Condition / Severity

3. Verify Event Triggers:

ET#	Description	Command
1	KAD06PBR, 22 CFCU low speed pushbutton	

SPECIAL INSTRUCTIONS

1. Provide operator marked up copy of S2.OP-ST.CBV-0003 for 22 CFCU.

INITIAL CONDITIONS

- Unit 2 is at 100% power.
- No major equipment out of service and no active Tech Specs are in effect.

INITIATING CUE

- You are the Reactor Operator.
- The CRS has directed you to **PERFORM** a scheduled surveillance test on 22 CFCU IAW S2.OP-ST.CBV-0003, Containment Systems – Cooling Systems.
- All Prerequisites are completed SAT and all required M&TE are installed in the field.
- A field NEO is standing by to provide SW header differential pressure when directed.
- Notify CRS of test results.

TASK STANDARD:

This task is satisfactorily met when the applicant has placed the CFCU in the required test condition, obtained the required test data, and determines 22 CFCU Test Result is UNSAT due to flow obtained being below the minimum flow for current delta P.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

ANSWER KEY

S2.OP-ST.CBV-0003(Q)

ATTACHMENT 2

22 CFCU DATA SHEET

1.0 **Reason For Test**

Date: Today	Post-Maintenance Operability Retest?	Yes	No ✓
	SAP Order Number(s): 72650412		

2.0 **Calibration Data**

Instrument/Test Equipment Description	Calibration Due Date	ID Number	Initials
2FA5495 22 CFCU Outlet Water Flow Indicator	6-22-23	27843	TP
Model HQS-2 Sensor Module 0-200 psi	8-14-23	61579	TP
Model HQS-2 Sensor Module 0-200 psi	8-14-23	61764	TP
Model PTE-1 Hand Held Heise Pressure Calibrator Gauge	9-22-23	62134	TP

3.0 **22 CFCU Performance Data**

Component	Test Data		Acceptance Criteria	Test Results SAT	Test Results UNSAT
22 CFCU	Start Time		22 CFCU operated for at least 15 minutes in LOW SPEED at ≥1465 gpm.	X	
22 CFCU	Stop Time				
22SW58	Cooling Water Flow Rate	1785			
22SW72					
22SW223					
21 SW HDR ΔP (78' IPA - 21 SW Piping Room)		106	Cooling Water Flow Rate is ≥ the Minimum Flow Rate obtained in Exhibit 1 for ΔP specified.		X
Minimum Flow Rate (Exhibit 1)		1811			

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
3.0		**PRECAUTIONS AND LIMITATIONS** 3.4 After recording Differential Pressure (DP) reading, select corresponding minimum flow value specified in Exhibit 1. • IF measured DP value is between DP values specified in Exhibit 1, THEN select minimum flow value for next higher DP value listed (e.g., if measured DP is 101 psid, use minimum flow required at 105 psid).	Operator reads and initials all P&Ls. **Evaluator's Note:** Understanding P&L 3.4 is important to properly determining which DP value to choose from Exhibit 1.	
Cue: Provide candidate MARKED UP surveillance procedure S2.OP-ST.CBV-0003. Examiners Note: Comments:				
4.0		**EQUIPMENT/MATERIAL REQUIRED**	**CUE**: Completed SAT by I&C Technicians.	
Cue: Examiners Note: Comments:				
5.1		CFCU Operability and Service Water Flow Verification		
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.1.1		PERFORM test on each Containment Fan Coil Unit required to be tested IAW the following instructions:		
Cue: Examiners Note: Comments:				
5.1.1.A		**ENSURE** CFCU being tested is in LOW SPEED IAW S2.OP-SO.CBV-0001(Q), Containment Ventilation Operation.	Operator places CFCU in LOW SPEED IAW S2.OP-SO.CBV-0001. Operator GOES TO S2.OP-SO.CBV-0001	
Cue: **Examiners Note**: **These next steps are from S2.OP-SO.CBV-0001, Containment Ventilation Operation.** Comments:				
5.1.2.A	*	IF shifting CFCU(s) from High Speed to Low Speed, THEN for each CFCU to be shifted: **PRESS** FAN HIGH SPEED STOP bezel.	Operator depresses HIGH SPEED STOP bezel for 22 CFCU and verifies STOP bezel illuminates.	
Cue: Examiners Note: Steps 5.1.2B and 5.1.2C should be coordinated for 25 second delay prior to energizing slow speed windings. Comments:				
5.1.2.B	*	**ENSURE** FAN STOP bezel is illuminated.	Operator verifies HIGH SPEED STOP bezel is illuminated.	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Cue: Examiners Note: Comments:				
5.1.2.C	*	When 25 seconds have elapsed from pressing HIGH SPEED STOP bezel, **PRESS** FAN LOW SPEED START bezel.	Operator waits 25 seconds from when HIGH SPEED STOP bezel was depressed. Operator depresses LOW SPEED START bezel for 22 CFCU and verifies START bezel illuminates.	
Cue: **Examiners Note**: **Simulator Operator**: ENSURE **ET-1** is TRUE when low speed start pushbutton is depressed. This will insert MALF: VL0570, 22SW223 fails to position set at 89**.** Comments:				
5.1.2.D		**ENSURE** following dampers are in indicated positions: ▪ ROUGH FLTR DAMPER CLOSED ▪ HEPA INLET DAMPER OPEN ▪ HEPA OUTLET DAMPER OPEN	Operator verifies the following for 22 CFCU: ▪ ROUGH FLT DAMPER CLOSED bezel extinguished. ▪ HEPA INLET DAMPER OPEN bezel illuminated. ▪ HEPA OUTLET DAMPER OPEN bezel illuminated.	
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
5.1.2.E		IF Service Water flow is less than 1465 gm, THEN: 1. STOP the CFCU 2. REFER to S2.OP-SO.SW-0005, SW System Operability Guidelines	Operator determines that SW flow is > 1465 gpm and marks step **N/A** and continues on.	
Cue: Examiners Note: Comments:				
5.1.2.F		**ENSURE** Service Water flow greater than or equal to 1465 gpm.	Operator verifies that SW flow is > 1465 gpm.	
Cue: Examiners Note: **Operator returns to S2.OP-ST.CBV-0003** Comments:				
5.1.1.B		**RECORD** Start Time in applicable Attachment(s), Section 3.0, for CFCU being tested.	Operator records START time.	
Cue: Examiners Note: Comments:				
ALTERNATE PATH STARTS HERE				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
4.1.11.3	*	WHEN at least 15 minutes has elapsed, THEN **RECORD** the following in applicable Attachment, Section 3.0, for CFCU being tested: • Stop Time • Cooling Water Flow Rate (gpm) • IF testing 21 OR 22 CFCU, THEN **RECORD** 21 SW HDR DP. • IF testing 23 CFCU, THEN **RECORD** both 21 SW HDR DP AND 22 SW HDR DP (Refer to Step 3.5). • IF testing 24 OR 25 CFCU, THEN **RECORD** 22 SW HDR DP. **RECORD** Test Results by initialing the SAT OR UNSAT column IAW Acceptance Criteria stated in the attachment.	Operator records STOP time. **Operator records SW flow on 22 CFCU (1785 +/- 5 gpm)** **Operator contacts field NEO and Records SW header DP of 106 psi** Operator determines that 23, 24, and 25 CFCUs are not being tested. **Operator records Test Results as UNSAT.** 22 CFCU is Inoperable based on Cooling Water Flow Rate is less than the Minimum Flow Rate (1811 gpm) required in Exhibit 1 selecting a SW DP of 110 psi (refer to P&L 3.4). P&L 3.3 requires CFCU SW flow rate ≥ 1465 gpm AND ≥ Min Flow Rate specified in Exhibit 1 to be Operable. **See Answer Key for completed Attachment 2, Section 3.0**	
Cue #1: Inform operator that 15 minutes have elapsed. **Cue #2**: When field operator is contacted; report 21 SW Header DP is reading 106 psi. **Examiners Note**: **Terminate the JPM when the operator Records Test Results as SAT or UNSAT.** Comments:				
Terminating Cue	**JPM COMPLETE**			

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-f **Revision No.:** 1

Task Title: Perform 22 CFCU Surveillance Test IAW S2.OP-ST.CBV-0003

Task No.: N0220130201

Source: Bank X New _____ Mod _____

System: 022 SF5 CCS Containment Cooling System

K/A Number / Description: A1.04 Ability to predict and/or monitor changes in parameters associated with operation of the Containment Cooling System, including: Cooling water flow

K/A Rating RO 3.3 SRO N/A

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: S2.OP-ST.CBV-0003

Reference(s): S2.OP-ST.CBV-0003 S2.OP-SO.CBV-0001

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Unit 2 is at 100% power.
- No major equipment out of service and no active Tech Specs are in effect.

INITIATING CUE

- You are the Reactor Operator.
- The CRS has directed you to **PERFORM** a scheduled surveillance test on 22 CFCU IAW S2.OP-ST.CBV-0003, Containment Systems – Cooling Systems.
- All Prerequisites are completed SAT and all required M&TE are installed in the field.
- A field NEO is standing by to provide SW header differential pressure when directed.
- Notify CRS of test results.

PSEG Nuclear LLC

Job Performance Measure

Respond to a failed high RCS T-avg channel and defeat the failed channel IAW S2.OP-AB.ROD-0003

JPM Number: 21-01 NRC Sim-g

Revision Number: 00

Date: 10/12/22

Developed By:	**K. Hantho** **Instructor**	**Date:**	10/12/22
Validated By:	**Gallagher/Saulsbery** **SME or Instructor**	**Date:**	11/21/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	10/12/22	New JPM

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

EG 1. Task description and number, JPM description and number are identified.

EG 2. Knowledge and Abilities (K/A) references are included.

EG 3. Performance location specified. (in-plant, control room, or simulator)

EG 4. Initial setup conditions are identified.

EG 5. Initiating and terminating cues are properly identified.

EG 6. Task standards identified and verified by SME review.

EG 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

EG 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-AB.ROD-0003 R23 Date Checked: 10/12/22

EG 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

Ed Gallagher	11-21-22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-256 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is at 100%.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Severity
RC0014a	21 HOT LEG RTD AVG SUMMATOR FAILS	NA	NA	NA	RT-1	630

SPECIAL INSTRUCTIONS

1. None.

INITIAL CONDITIONS

- Salem unit 2 is at 100%
- No major equipment is out of service and no Tech Specs are active.

INITIATING CUE

- You are the Reactor Operator.
- Respond to all alarms and indications.
- Your evaluator will respond to all alarms not related to your task.

TASK STANDARD:

The task is satisfactorily met when the applicant has taken the Rod control system to manual, restored Tavg, stabilized PZR level and defeated the failed Tavg channel.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Insert RT-1 Examiners Note: Operator recognizes unwarranted rod motion and takes rod control to manual Comments:				
N/A	*	Operator attempts to identify failure, and will take action to stop rod insertion if not evident immediately by verifying no runback in progress	Operator verifies no runback in progress and takes rods to manual.	
Cue: Examiners Note: Comments:				
3.1		Enter S2.OP-AB.ROD-0003 and Place the rod bank selector switch in Manual	Enters AB.ROD-3 and Places the Rod control bank selector switch in Manual.	
Cue: Examiners Note: Comments:				
3.2		Did rod motion stop	Operator states Yes and continues to step 3.4	
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		ALTERNATE PATH STARTS HERE		
3.4		**MONITOR** Tavg for necessary adjustments until the cause of the transient is known and the Plant is stable.	Operator will initial and monitor T_{avg}	
Cue: Examiners Note: Comments:				
3.5	*	**ADJUST** Rods in manual to control Tavg within 1.5°F of T_{avg} program. (Attachment 1, Tavg Program VS % Reactor Power)	Operator will verify within 1.5 and continue or adjust to with 1.5 ° F by withdrawing rods and then move on.	
Cue: If asked CRS directs you to restore Rods to ≤ 1 ° F **Examiners Note**: Tavg may be within 1.5°F depending on how quickly rods were placed in manual. Comments:				
3.6		Has a NIS channel Failed High?	Operator verbalizes and initials NO and goes to step 3.8	
Cue : Examiners Note: Comments:				
3.8		**STOP** any dilution of the Reactor Coolant System.	Operator verifies makeup system Out of service and states no dilution in progress	
Cue: Examiners Note: Comments:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
3.9		Has a T_{avg} channel failed high?	Answers Yes and goes to step 3.10	
Cue: Examiners Note: Comments:				
3.10		**RETURN** Pressurizer level to program as follows: (Attachment 2, Pressurizer Level Program) ___ A. **PLACE** the Charging System Master Flow Controller in MANUAL. ___ B. **ADJUST** Charging System Master Flow Controller to restore Pressurizer level to program.	Places Master flow controller or CV55 in manual and adjusts to restore level to program band	
Cue: Examiners Note: The PZR level will remain at 59% as the control signal is capped at the 100% value Comments:				
3.11	*	**SELECT** DEVIATION DEFEAT for the affected LOOP at the following 2CC2 bezel sections: ♦ REACTOR COOLANT DIFFERENTIAL TEMPERATURES ♦ REACTOR COOLANT AVERAGE TEMPERATURES	Selects deviation defeat on 2CC2 for channel 1 Differential temperature and average temperature.	
Cue: Examiners Note: Comments:				
3.12	*	**SELECT** a valid Channel on the Recorder Select at the Control Console for the following: ♦ REACTOR COOLANT DIFFERENTIAL TEMPERATURES ♦ REACTOR COOLANT AVERAGE TEMPERATURES	Selects recorders to Channel 2, 3 or 4 for Differential temperature and average temperature.	
Cue: Examiners Note:				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Comments:				
3.13		When pressurizer level has been restored to program, **RETURN** the Charging System Master Flow Controller in AUTO.	Verifies level is at program and returns charging master flow controller to Auto	
Cue: If desired, may walkthrough remaining steps or provide cue that level has been returned to program Examiners Note: depending on level error this step may take a while to restore Comments:				
3.14		Are control rods above Rod Insertion Limits?	Verifies Rods are above insertion limits	
Cue: Examiners Note: Comments:				
3.16		IF manual rod control is NOT required, THEN **PLACE** the Rod Control System in automatic control.	Places rod control in Auto	
Cue: CRS directs rods restored to auto control Examiners Note: May restore the rods to the all rods out position before returning to AUTO Comments:				
Cue: Examiners Note: Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-g **Revision No.:** 00

Task Title: Respond to a failed high RCS T-avg channel and defeat the failed channel IAW S2.OP-AB.ROD-0003

Task No.: N1150090501

Source: Bank _____ New X Mod _____

System: 012 RPS Reactor Protection System

K/A Number / Description: A4.05 Ability to manually operate and/or monitor in the control room: Channel defeat controls

K/A Rating RO 3.8 SRO N/A

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **6** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: S2.OP-AB.ROD-0003

Reference(s): S2.OP-AB.ROD-0003

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Salem unit 2 is at 100%
- No major equipment is out of service and no Tech Specs are active.

INITIATING CUE

- You are the Reactor Operator.
- Respond to all alarms and indications.
- Your evaluator will respond to all alarms not related to your task.

PSEG Nuclear LLC

Job Performance Measure

PERFORM A CONTAINMENT PRESSURE RELIEF

JPM Number: 21-01 NRC Sim-h

Revision Number: 0

Date: 07/26/2022

Developed By:	**R. Chan** **Instructor**	**Date:**	7/26/22
Validated By:	**Westlake / Militti** **SME or Instructor**	**Date:**	8/19/22
Reviewed By:	**M. Winkelspecht** **Operations Representative**	**Date:**	11/29/22
Approved By:	**M. Wadusky** **Training Department (Print/Sign)**	**Date:**	11/29/22

REVISION RECORD (Summary)

Revision Number	Date	Reason
00	7/26/22	Converted to new JPM template.

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

NOTE

All steps of this checklist should be performed upon initial validation. Prior to JPM usage, revalidate JPM using steps 8 and 11 below.

RC 1. Task description and number, JPM description and number are identified.

RC 2. Knowledge and Abilities (K/A) references are included.

RC 3. Performance location specified. (in-plant, control room, or simulator)

RC 4. Initial setup conditions are identified.

RC 5. Initiating and terminating cues are properly identified.

RC 6. Task standards identified and verified by SME review.

RC 7. Critical steps meet the criteria for critical steps and are identified with an asterisk (*)

RC 8. Verify the procedure referenced by this JPM matches the most current revision of that procedure:

Procedure(s) S2.OP-SO.CBV-0002 Rev. 18
S2.OP-AB.RAD-0001 Rev. 30
S2.OP-AR.ZZ-0001 Rev. 60

Date Checked: 7/21/22

RC 9. Pilot test the JPM:

a. Verify cues both verbal and visual are free of conflict, and

b. Ensure performance time is accurate.

NA 10. If the JPM cannot be performed as written with proper responses, then revise the JPM.

NA 11. When JPM is revalidated, SME or instructor sign and date JPM cover page

R. Chan	11/28/22
SME/Instructor	Date
SME/Instructor	Date
SME/Instructor	Date

SIMULATOR SETUP INSTUCTIONS

- **RESET simulator to IC-257 and VERIFY the below events on the Instructor Station**.
- Salem Unit 2 is at 100% power.
- No major equipment are out of service and no Tech Specs are active.
- The following malfunctions, overrides, and remotes were required to develop this IC.

MALF ID #	Description	Delay Time	Initial Value	Ramp Time	Trigger	Final Value
01	VL0282 2VC6 fails to position(0-100%)	N/A	N/A	N/A	ET-1	100
02	VL0283 2VC5 fails to position(0-100%)	N/A	N/A	N/A	ET-1	100
03	RM0203C Process Rad mon 2R41A fails anywhere	20 sec	Current	1 minute	ET-3	300
04	VL0284 2CBV36 fails to position(0-100%)	5 sec	N/A	N/A	ET-3	100

OVR	Description	Assigned value
01	AB14 A1 LO	ON

ET #	Description	Assigned malfunction
1	KAB13PNR//2VC6 CNTMNT pressure vacuum reli	VL0282
2	KAB14PNR//2VC5 CNTMNT pressure vacuum reli	VL0283
3	KAB14ANR//CNTMNT pressure relief damper open	RM0203C
4	KAB13TCR//2VC6 CNTMNT pressure vacuum reli Optional command DMF VL0282	DMF VL0282
5	KAB14TCR//2VC5 CNTMNT pressure vacuum reli optional command DMF VL0283	DMF VL0283
6	KAB14DCR//CNTMT pressure relief damp clos	DMF VL0284
7	KAB14DCR//CNTMT pressure relief damp clos	DOR QAB14ARR

SPECIAL INSTRUCTIONS

1. S2.OP-SO.CBV-0002 is MARKED UP in the simulator book with all sections completed up to section 5.2.

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- Preparations to perform a containment pressure relief per Section 5.1 of S2.OP-SO.CBV-0002, Containment Pressure – Vacuum Relief System Operation, are complete.

INITIATING CUE

The CRS directs you to perform a containment pressure relief per Section 5.2 of S2.OP-SO.CBV-0002.

TASK STANDARD:

The task is satisfactorily met when the applicant has placed containment pressure relief in service, and upon a high radiation alarm, manually closes 2VC5 and 2VC6 valves to isolate the radiological release from containment.

****

Information for Evaluators Use:

UNSAT requires written comments on the respective step.

(*) **Denotes critical steps**

If Time Critical, estimated time is the Time Critical time.

The comment section should be used to document the reason that a step is marked as unsatisfactory and to document unsatisfactory performance relating to management expectations.

Some operations that are performed from outside of the control room may require multiple steps. These items may be listed as individual steps in this JPM. It is acceptable for the candidate to direct the local operator to perform groups of procedure steps instead of calling for each individual item to be performed.

The time clock starts when the candidate acknowledges the initiating cue.

RECORD JPM Start Time:_____

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
N/A		**RECORD** the JPM Start Time when the operator acknowledges READY TO START JPM.		N/A
Cue: Provide candidate MARKED UP copy of S2.OP-SO.CBV-0002. Examiners Note: None. Comments: N/A				
5.2.2		**RECORD** the following Release Data on Attachment 2, Section 1.0: ♦ Pressure Relief Start ♦ Initial Containment Pressure ♦ Initial reading on available radiation monitors 2R41D (preferred) and 2R12A	Operator records the required data.	
Cue: None. Examiners Note: None Comments:				
5.2.3.A		**INITIATE** Containment Relief as follows: **MONITOR** available radiation monitors 2R41D (preferred) and 2R12A.	Operator can setup a trend using the Plant Computer OR monitor the Radiation Monitors on 2RP1.	
Cue: None. **Examiners Note**: 2R12A and 2R41D are located on 2RP1 Panel. Comments:				
5.2.3.B	*	IF Containment pressure is <0.5 psig, THEN: 1. **OPEN** 2VC6, ISOL VLV	Operator confirms containment pressure is less than 0.5 psig by reading the containment pressure indicators on 2CC1.	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		2. **OPEN** 2VC5, ISOL VLV 3. **OPEN** PRESSURE RELIEF DAMPER	Operator depresses the OPEN bezels for 2VC6, 2VC5, and the PRESSURE RELIEF DAMPER on 2CC1 and verifies the Red OPEN bezels are LIT.	
Cue: None Examiners Note: None Comments:				
5.2.3.C		Step is N/A		
Cue: None. Examiners Note: None Comments:				
5.2.4		**RECORD** time 2VC5 and 2VC6 are OPENED in Control Room Narrative Log IAW OP-SA-470-1001, CYCLIC Data Monitoring Program.		
Cue: The extra NCO will record the open time for 2VC5 and 2VC6. Examiners Note: None Comments:				
ALTERNATE PATH STARTS HERE				
Cue: SIMULATOR OPERATOR ensure ET-3 is true and MALF 03 RM0203C is ramping in. This will cause the 2R41D Rad Monitor channel to go into Alarm, actuate OHA A-6 RMS HI RAD OR TRBL, previous malfunctions have been enabled to prevent the automatic closures of the pressure relief dampers, 2VC5 and 2VC6 valves on a high radiation alarm.				

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
Examiners Note: The candidate may take the action per step 5.2.5 of S2.OP-SO.CBV-0002, **OR** refer to the OHA A-6 and then take the actions per S2.OP-AB.RAD-0001, Abnormal Radiation, to manually close the dampers per Attachment 10, both paths are acceptable. Comments:				
ACTIONS PER S2.OP-SO.CBV-0002				
5.2.5	*	When Containment pressure decrease to required value: A. **CLOSE** PRESSURE RELIEF DAMPER B. **CLOSE** 2VC6, ISOL VLV C. **CLOSE** 2VC5, ISOL VLV	Operator depresses the CLOSE bezel pushbuttons for PRESSURE RELIEF DAMPER, 2VC6, and 2VC5 and verifies the Green bezel is LIT.	
Cue: None **Examiners Note**: The candidate is required to close 2VC5 and 2VC6 to satisfy the Task Standard. Comments:				
ACTIONS PER S2.OP-AB.RAD-0001				
3.1		**DETERMINE** affected Radiation monitor.		
Cue: None Examiners Note: None Comments:				
3.2		Is the alarm, warning, or rising indication valid? (**EVALUATE** the following): ♦ Chemistry Sampling ♦ Rising activity on related Radiation Monitors ♦ Maintenance activities in vicinity of radiation detector	Operator determines the alarm is VALID.	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		♦ Transfer of radioactive resins or other materials in affected area ♦ No indication of failure or fault on affected Radiation Monitor ♦ Radioactive release from Salem or Hope Creek		
Cue: IF operator questions if alarm is VALID or Not, THEN state: CRS directs continuing with procedure since Rad Monitor does not appear to be failed. Examiners Note: None Comments:				
3.3		**ANNOUNCE** the following on Plant PA system: ♦ Affected Radiation Monitor number and name ♦ Location of Radiation Monitor area with elevated indication	Operator makes page announcement	
Cue: None Examiners Note: None Comments:				
3.4		IF a release to the environment is in progress, THEN **DIRECT** Shift Radiation Protection Technician to complete and provide Page 2 of the Station Status Checklist (SSCL) for release rate determination.	Operator notifies Rad Pro of actions.	
Cue: CRS will direct Radiation Protection to perform Station Status Checklist. Examiners Note: None Comments:				
3.5		**NOTIFY** SM/CRS to REFER TO Event Classification Guide,	Notifies CRS of required actions.	

STEP	CRITICAL	ELEMENT	STANDARD	GRADE (S/U)
		ODCM and Technical Specifications.		
Cue: CRS will refer to ECG, Tech Specs, and ODCM. Examiners Note: None Comments:				
3.5		**INITIATE** the applicable attachment for affected Radiation Monitor:	Operator initiates Attachment 1 Step 10 based on 2R41D in alarm.	
Cue: None. Examiners Note: None Comments:				
10.0		IF 2R41D, Plant Vent Release Rate is the affected monitor, THEN:		
Cue: None Examiners Note: None Comments:				
10.1	*	**ENSURE** 2VC1, 2VC4, 2VC5 AND 2VC6 are CLOSED.	Operator depresses the CLOSE bezel pushbuttons for 2VC5 and 2VC6 and verifies Green bezel is LIT.	
Cue: None **Examiners Note**: Only 2VC5 and 2VC6 are required to be CLOSED to satisfy the Task Standard. Comments:				
Terminating Cue		**JPM COMPLETE**		

RECORD JPM Stop Time: _____

Operator's Name: _____ **Job Title:** RO _____ SRO _____

Facility: Salem **JPM No.:** 21-01 NRC Sim-h **Revision No.:** 0

Task Title: Perform a Containment Pressure Relief

Task No.: N0220080101

Source:
Bank _____ New X Mod _____

System: 029 Containment Purge System (SF8)

K/A Number / Description: A2.02 Ability to (a) predict the impacts of the following on the Containment Purge System and (b) based on those predictions,, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Adverse environmental conditions affecting radioactive release

K/A Rating RO 3.0 SRO 3.0

Task Applicability: SRO Only _____ RO/SRO X AO/RO/SRO _____ Other _____

Time-Critical: Yes _____ No X **Alternate Path:** Yes X No _____

Estimated Time to Complete: **10** Minutes **Actual Time Used:** Minutes

Method of Testing: Simulated Performance _____ Actual Performance X

Location: Classroom _____ Simulator X In-Plant _____ RCA _____

Required Materials: S2.OP-SO.CBV-0002

Reference(s): S2.OP-SO.CBV-0002 S2.OP-AB.RAD-0001 S2.OP-AR.ZZ-0001

EVALUATION SUMMARY:

Were all the Critical Elements (steps) performed satisfactorily? Yes ____ No ____

The operator's performance was evaluated against the standards contained in this JPM, and has been determined to be: Satisfactory _____ Unsatisfactory _____

Comments:

Evaluator's Name: _____

Evaluator's Signature: _____ **Date:** _____

STUDENT HANDOUT

INITIAL CONDITIONS

- Salem Unit 2 is at 100% power
- Preparations to perform a containment pressure relief per Section 5.1 of S2.OP-SO.CBV-0002, Containment Pressure – Vacuum Relief System Operation, are complete.

INITIATING CUE

The CRS directs you to perform a containment pressure relief per Section 5.2 of S2.OP-SO.CBV-0002.