WRITTEN / ORAL / ONLINE EXAMINATION KEY COVER SHEET

Examination Number/Title:	**RO NRC Exam**
Training Program:	**PBN LOI Initial License Training Program**
Course/Lesson Plan Number(s):	**RO NRC EXAM**

Total Points Possible: 75.00	**PASS CRITERIA: ≥ 80%**	**Exam Time:**

	Yes	No		Yes	No
This is an alternate examination; verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☐	This is a remediation exam. Verified the questions are different from the failed exam by at least the following criteria listed below: • 70% for Maintenance/Technical • 90% for Operations training programs	☐	☐
This is an initial training examination; verified at least 30% of the questions are different from previous administration of the same exam. This requirement is NA if ≥ 2 years since previous administration of the exam.	☐	☐	This is a LOCT annual operating exam or biennial comprehensive remedial exam, verified the questions are 100% different from the failed exam.	☐	☐
This is a non-randomly generated exam from an electronic exam bank, printed out or administered online. Verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☐	This is a randomly generated exam from an electronic exam bank, printed out or administered online. Verified the exam bank has 3 questions per objective if one test item on exam for the objective. If 2 or more test items on exam for an objective, then 6 questions are in bank.	☐	☐

NOTE:
• See TR-AA-230-1003, SAT Development, for exam development and review guidelines.
• NRC exams may require additional information. Refer to fleet and site specific procedures.
			Key should contain the following:
			• Learning Objective Number
			• Test Item
			  • Question or Statement
			  • All possible answers
			  • Correct Answer Indicated
			  • Point Value
			  • References (if applicable)		

EXAMINATION REVIEW AND APPROVAL:	
Developed by:	**Date:**
Instructional Review of Written Exam (Qualified Instructor):	**Date:**
Technical Review (SME):	**Date:**
Approved by Training Supervisor:	**Date:**
Approved by Training Program Owner (or line designee):	**Date:**

Indicate in the following table if any changes are made to the exam after approval:

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR# (if applicable)	PREPARER / SUPERVISOR	DATE / DATE

POINT BEACH NUCLEAR PLANT EXAM KEY
Test ID: RO NRC Exam Test Form: 0
Test File: RO NRC EXAM

#	ID	Points	Type	Answers 0
1	2017 NRC 1	1.00	MCS	C
2	2017 NRC 2	1.00	MCS	B
3	2017 NRC 3	1.00	MCS	D
4	2017 NRC 4	1.00	MCS	C
5	2017 NRC 5	1.00	MCS	D
6	2017 NRC 6	1.00	MCS	C
7	2017 NRC 7	1.00	MCS	A
8	2017 NRC 8	1.00	MCS	C
9	2017 NRC 9	1.00	MCS	A
10	2017 NRC 10	1.00	MCS	A
11	2017 NRC 11	1.00	MCS	D
12	2017 NRC 12	1.00	MCS	C
13	2017 NRC 13	1.00	MCS	A
14	2017 NRC 14	1.00	MCS	D
15	2017 NRC 15	1.00	MCS	B
16	2017 NRC 16	1.00	MCS	C
17	2017 NRC 17	1.00	MCS	C
18	2017 NRC 18	1.00	MCS	B
19	2017 NRC 19	1.00	MCS	B
20	2017 NRC 20	1.00	MCS	D
21	2017 NRC 21	1.00	MCS	D
22	2017 NRC 22	1.00	MCS	A
23	2017 NRC 23	1.00	MCS	D
24	2017 NRC 24	1.00	MCS	D
25	2017 NRC 25	1.00	MCS	A
26	2017 NRC 26	1.00	MCS	A
27	2017 NRC 27	1.00	MCS	D
28	2017 NRC 28	1.00	MCS	B
29	2017 NRC 29	1.00	MCS	B
30	2017 NRC 30	1.00	MCS	B
31	2017 NRC 31	1.00	MCS	D
32	2017 NRC 32	1.00	MCS	A
33	2017 NRC 33	1.00	MCS	D
34	2017 NRC 34	1.00	MCS	A
35	2017 NRC 35	1.00	MCS	C
36	2017 NRC 36	1.00	MCS	A
37	2017 NRC 37	1.00	MCS	C
38	2017 NRC 38	1.00	MCS	D
39	2017 NRC 39	1.00	MCS	B
40	2017 NRC 40	1.00	MCS	B
41	2017 NRC 41	1.00	MCS	A
42	2017 NRC 42	1.00	MCS	A
43	2017 NRC 43	1.00	MCS	A
44	2017 NRC 44	1.00	MCS	C
45	2017 NRC 45	1.00	MCS	C

#	ID	Points	Type	Answers 0
46	2017 NRC 46	1.00	MCS	D
47	2017 NRC 47	1.00	MCS	B
48	2017 NRC 48	1.00	MCS	B
49	2017 NRC 49	1.00	MCS	C
50	2017 NRC 50	1.00	MCS	B
51	2017 NRC 51	1.00	MCS	D
52	2017 NRC 52	1.00	MCS	C
53	2017 NRC 53	1.00	MCS	A
54	2017 NRC 54	1.00	MCS	B
55	2017 NRC 55	1.00	MCS	A
56	2017 NRC 56	1.00	MCS	D
57	2017 NRC 57	1.00	MCS	C
58	2017 NRC 58	1.00	MCS	A
59	2017 NRC 59	1.00	MCS	C
60	2017 NRC 60	1.00	MCS	C
61	2017 NRC 61	1.00	MCS	B
62	2017 NRC 62	1.00	MCS	A
63	2017 NRC 63	1.00	MCS	B
64	2017 NRC 64	1.00	MCS	D
65	2017 NRC 65	1.00	MCS	C
66	2017 NRC 66	1.00	MCS	A
67	2017 NRC 67	1.00	MCS	B
68	2017 NRC 68	1.00	MCS	B
69	2017 NRC 69	1.00	MCS	A
70	2017 NRC 70	1.00	MCS	D
71	2017 NRC 71	1.00	MCS	A
72	2017 NRC 72	1.00	MCS	C
73	2017 NRC 73	1.00	MCS	C
74	2017 NRC 74	1.00	MCS	A
75	2017 NRC 75	1.00	MCS	B

1. 2017 NRC 001/EPE/007EK2.03/3.5/2-RI/RO/BANK/883D195 SH13/053.02.LP0361.004
Given the following:

- Unit 1 is at 7% of Rated Thermal Power

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

A. Four CLNT FLOW LOOP bistables tripped

B. Three PRZR LO PRESS bistables tripped

C.✓ Two PRZR HI PRESS bistables tripped

D. One SG LO LEVEL bistable tripped

RO Tier 1 Group 1

Source: Bank

Question History:
2009 Millstone NRC RO 1

K/A:
007EK2.03 Reactor Trip
Knowledge of the interrelations between a reactor trip and the following:
Reactor trip status panel. (Imp 3.5/3.6)

Cognitive Level:
Comprehension 2-RI: The operator must understand the initial conditions, realize the status of the permissives, and determine which of the choices will cause a reactor trip based on those permissives.

Reference:
883D195 Sh 13, Pressurizer Trip Signals Logic Diagram, Rev 6
883D195 Sh 14, Steam Generator Trip Signals Logic Diagram, Rev 10
883D195 Sh 15, RC Trip Signal Logic Diagram, Rev 9

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the power level, permissives P-7 (10%), P-8 (35%) are not met, AND the trip logic (2 of 3) is met, therefore PRZR HI PRESS bistables will have caused the reactor trip.

A **INCORRECT:** Plausible due to 4 bistables are said to be tripped, with no distinction between loops. Given that information, 3 from one loop, or 2 from each loop may be tripped, so both the single loop and two loop low flows conditions can exist simultaneously. Both signals would produce a reactor trip if the correct permissive was met.

B **INCORRECT:** Plausible as this is in excess of the trip logic, only 2 of 4 needed, and this is blocked until 10% power.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible because 2 of 3 low-low SG water levels are needed to cause a trip, and only one low level when combined with a steam flow feed flow mismatch is needed, neither of which is dependent on permissives being met.

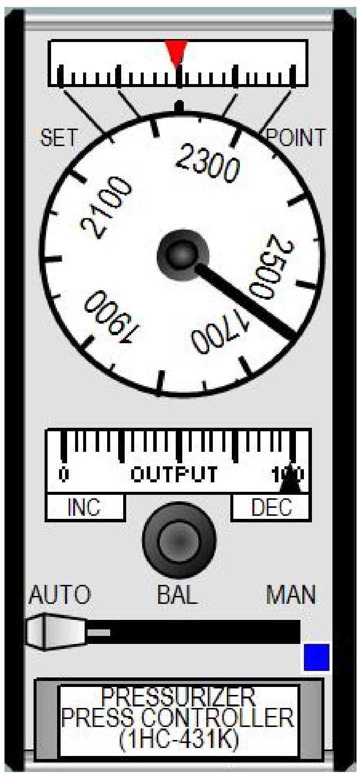
Learning Objective:

DESCRIBE the Reactor Trips, automatic functions, and interlocks for the Reactor Protection System. Description should include actuation setpoints, actuation logic, logic acceptability, requirements to enable actuation, and protection afforded by each of the signals/processes.
(053.02.LP0361.004)

2. 2017 NRC 002/APE/008AK2.03/2.5/2-RI/RO/NEW/883D195 SH 18/055.03.LP2438.001

Given the following:

- Unit 1 is at Rated Thermal Power
- 1PT-449, Pressurizer Pressure Transmitter failed HIGH
- PRIOR to the instrument being removed from service, 1HC-431K, Pressurizer Press Controller fails as pictured below:



Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "no ARB or SOP actions have been taken."

What is the impact on the following?

	1RC-431A/1RC-431B Pressurizer Spray valves	1RC-431C Pressurizer PORV
A.	OPEN	SHUT
B.✓	OPEN	OPEN
C.	SHUT	SHUT
D.	SHUT	OPEN

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
008AK2.03 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: **Controllers and positioners.** (Imp 2.5/2.4)

Cognitive Level:
Comprehension 2-RI: The operator must understand the initial condition; determine what the impact of the failed instrument as well as the failed controller output will be on the pressurizer spray and PORVs.

Reference:
883D195 Sh 18, Pressurizer Pressure and Level Control Logic Diagram, Rev 13
STPT 5.3 Pressurizer Pressure and Level Control Setpoint Document, Rev 11 Sections 1.1 and 1.1.3

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions, PT-449 affects 1RC-431C, providing one half of the signal required to open it. Given the controller failure, the student must recall the output signal will cause the spray valves to open, and also provide the second input to 1RC-431C causing it to open.

A **INCORRECT:** Sprays are correct and PORV is incorrect. Plausible if the student does not recall the inputs to 1RC-431C.

B **CORRECT:** See above explanation.

C **INCORRECT:** Both the sprays and PORV are incorrect. Plausible if the student has the inputs signals to the PORV incorrect and the output of the controller backwards, and thinks it will cause the heaters to energize raising pressure.

D **INCORRECT:** Sprays are incorrect, and PORV is correct. Plausible if the student has the output of the controller backwards, and thinks it will cause the heaters to energize raising pressure until a PORV lifts.

Learning Objective:

DESCRIBE the plant and operator(s) response to the following conditions:

a. Failure of Pressurizer Pressure and/or Level Control system
b. Reactor Coolant System leak
c. Reactor Coolant Pump malfunction
d. Steam Generator Tube leak

(055.03.LP2438.001)

3. 2017 NRC 003/EPE/009EK3.28/4.5/2-DR/RO/NEW/AOP-1A/055.03.LP2438.004

Given the following:

- Unit 1 is at 75% of Rated Thermal Power
- A reactor coolant leak has occurred
- The crew is implementing AOP-1A, Reactor Coolant Leak, and is currently carrying out the diagnostic steps
- Plant conditions are:
    - RCS Pressure is 2245 psig and STABLE
    - TAVE 567°F and STABLE
    - Pressurizer Level 27.8% and LOWERING SLOWLY
    - VCT Level is 25% and LOWERING SLOWLY

What are the required actions based on the plant conditions?

A. Manually trip the reactor and stabilize the plant only

B. Monitor the plant and continue with diagnostic steps

C. Isolate Letdown and continue with diagnostic steps

D.✓ Manually trip the reactor and manually initiate Safety Injection and Containment Isolation

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
009EK3.28 Small Break LOCA
Knowledge of the reasons for the following responses as they apply to the small break LOCA: **Manual ESFAS initiation requirements.** (Imp 4.5/4.5)

Cognitive Level:
Comprehension 2-DR: The operator must understand the initial condition; determine if the plant conditions are adequate for the current power level and then determine what the necessary actions are.

Reference:
AOP-1A Unit 1, Reactor Coolant Leak, Rev 18, Step 1
STPT 5.3, Pressurizer Pressure and Level Control Setpoint Document, Rev 11, Section 1.2

Proposed reference to be provided to the applicants during examination:
None

Justification:
Given the initial conditions, of 75% of RTP, the student must determine that pressurizer level is greater than 10% from program, and therefore a reactor trip is required.

A **INCORRECT:** Plausible because only a reactor trip is required if the charging pump suction shifts from the VCT to the RWST and the VCT level is slowly lowering, but no impairment to the makeup system exists.

B **INCORRECT:** Plausible if the student has a misconception on the pressurizer level requirement, as the rest of the plant conditions require no actions other than isolating of Letdown.

C **INCORRECT:** Plausible due to the lowering of the pressurizer level, without an explicit bullet stating charging flow was adjusted to maintain pressurizer level.

D **CORRECT:** See above explanation.

Learning Objective:
Given access to the Site Specific Simulator or specific plant conditions, RESPOND to the following events:
a. Reactor Coolant System leakage
b. Reactor Coolant Pump malfunctions
c. Steam Generator Tube leak
(055.03.LP2438.004)

4. 2017 NRC 004/EPE/011EK3.06/4.3/3-SPK/RO/NEW/883D195 SH 7/053.06.LP0486.014

Given the following:

- Unit 1 is cooling down in preparation for a Refueling Outage
- RCS temperature is 375°F
- Pressurizer Level is 70%
- Pressurizer Pressure is 1000 psig
- Safety Injection has been blocked
- A Large Break LOCA occurs

For this event, which of the following signals would result in a Containment Isolation signal?

A. Manual CI only

B. Manual SI and Manual CI only

C.✓ High Containment Pressure SI and Manual CI only

D. Low Pressurizer Pressure SI and Manual CI only

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
011EK3.06 Large Break LOCA
Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: **Actuation of Phase A and Phase B during LOCA Initiation.**
(Imp 4.3/4.3)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions of the event, and apply the effects of the Large Break LOCA to those conditions.

Reference:
883D195 Sh 7, Safeguards Actuation Signals Logic Diagrams, Rev 25

Proposed reference to be provided to the applicants during examination:
None

Justification:

With the automatic Safety Injection signal blocked, steamline break outside containment, Steam Generator Tube Rupture, and small break LOCAs would not result in an automatic Containment Isolation signal. The automatic SI from high containment pressure is not blocked, so that would actuate on a LBLOCA and cause Containment Isolation. The manual CI is not blocked.

A **INCORRECT:** Manual CI would provide a CI signal, but so would Auto SI due to High Containment Pressure

B **INCORRECT:** Manual CI would provide a CI signal, but Manual SI does not

C **CORRECT:** See above explanation.

D **INCORRECT:** Manual CI would provide a CI signal, but Low Pressurizer Pressure SI has been blocked.

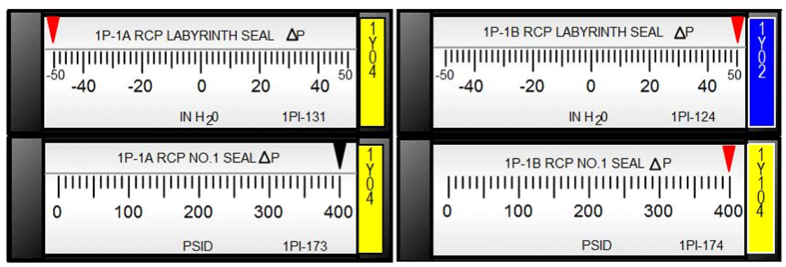
Learning Objective:

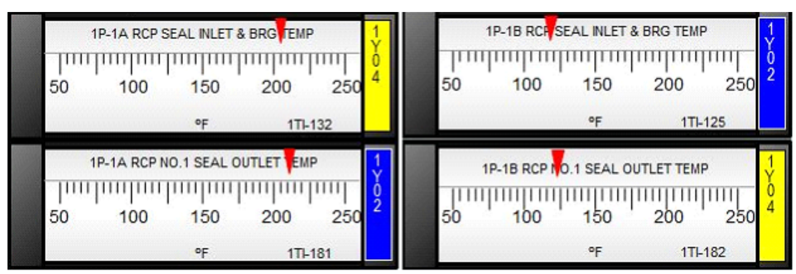
DESCRIBE the automatic functions associated with the Engineered Safety Features Actuation System and its major components. Description should include actuation setpoints, actuation logic, logic cceptability, requirements to enable actuation,and protection afforded by the system.
(053.06.LP0486.014)

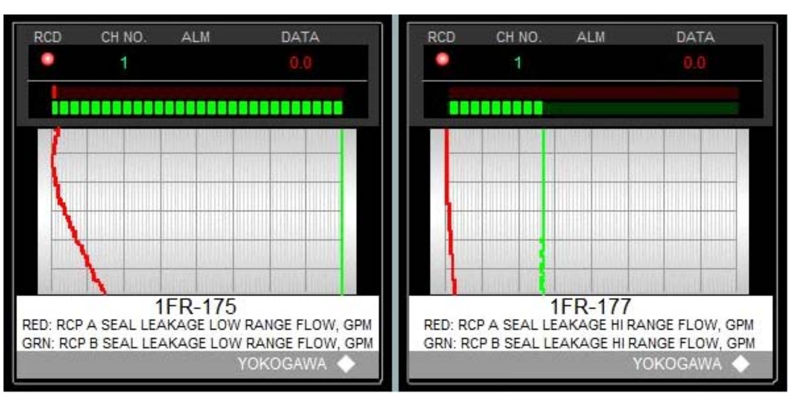
5. 2017 NRC 005/APE/015/017AA1.22/4.0/3-SPR/RO/BANK/AOP-1B/055.03.LP2438.001

Given the following:

- Unit 1 is at Rated Thermal Power
- The following annunciators on 1C03 1D are LIT
  - 1-1, 1P-1A OR B RCP LABYR SEAL WATER INLET OR BEARING TEMP HIGH
  - 1-2, 1P-1A RCP STAND PIPE LEVEL HIGH
  - 2-1, 1P-1A OR B RCP LABYR SEAL ΔP LOW
  - 3-1, 1P-1A OR B RCP NO. 1 SEAL WATER OUTLET TEMPERATURE HIGH
  - 3-2, 1P-1A RCP NO. 1 SEAL WATER FLOW HIGH OR LOW
- RDCT level is rising approximately 10% per minute
- RCP indication on 1C04 are as follows:







Question continued on next page

5. 2017 NRC 005/APE/015/017AA1.22/4.0/3-SPR/RO/BANK/AOP-1B/055.03.LP2438.001

Question continued from previous page

What is the status of the ‘A’ RCP seal package?

A. #1 seal is failing only

B. #2 seal is failing only

C. #3 seal is failing only

D.✓ #1 and #2 seals are failing

RO Tier 1 Group 1

Source: Bank

Question History:
2009 Millstone NRC SRO 76

K/A:
015/017AA1.22 Reactor Coolant Pump (RCP) Malfunctions
Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): **RCP seal failure/malfunction.** (Imp 4.0/4.2)

Cognitive Level:
Comprehension 3-SPR: The student must diagnose the seal package failure based on the plant indications.

Reference:
AOP-1B Unit 1, Reactor Coolant Pump Malfunction, Rev 24, Step Steps 4, 8,and 10, and Foldout Page Item 3
BG AOP-1D, Reactor Colant Pump Malfunction Background Document, Rev 14, Steps 4, 8,and 10, and Foldout Page Item 3

Proposed reference to be provided to the applicants during examination:
None

Justification:

The #1 and #2 seals have failed. The #1 seal is indicated by an increase in temperature on the RCP seal inlet and bearing temperature as well as the RCP No 1 seal outlet temp and the decrease in lab seal ΔP. The temperatures would not increase if only #2 seal failed, nor would the lab seal ΔP decrease. Also the annunciators 1-1, 2-1, 3-1, and 3-2 are indicative of a #1 seal failure, while 1-2 and 3-2 are indicative of a #2 seal failure. The low seal leakage is also a symptom of a #2 seal failure. So combining the symptoms, both seals #1 and #2 have failed.

A **INCORRECT:** Plausible as the indication of an increase in temperatures, decrease in lab seal ΔP are symptoms of a #1 seal failure, it does not explain the additional annunciators nor the low leakoff valves.

B **INCORRECT:** Plausible as the indication of a lowering seal leakage, as well as annunciators 1-2 and 3-2 are symptoms of a #2 seal failure, it does not explain the additional annunciators nor the increased temperatures or decreased ΔP.

C **INCORRECT:** Plausible as the stand pipe and leakoff symptoms could be confused with this failure as it is neither Seal #1 only or Seal #2 only.

D **CORRECT:** See above explanation.

Learning Objective:

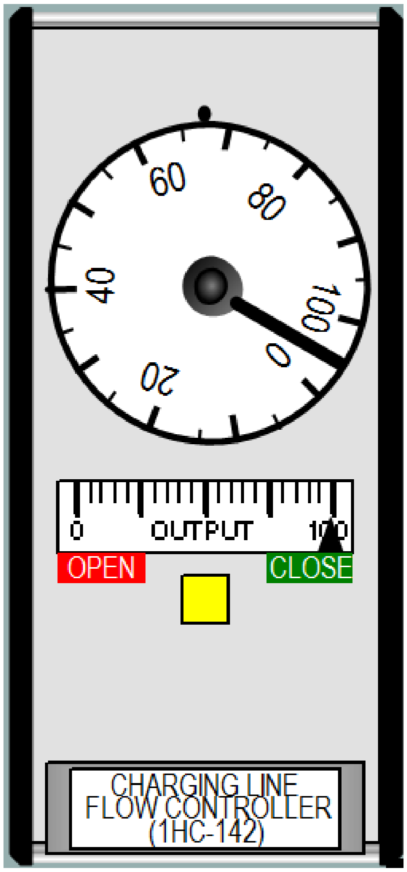
DESCRIBE the plant and operator(s) response to the following conditions:

a. Failure of Pressurizer Pressure and/or Level Control system
b. Reactor Coolant System leak
c. Reactor Coolant Pump malfunction
d. Steam Generator Tube leak

(055.03.LP2438.001)

6. 2017 NRC 006/APE/022AA1.02/3.0/2-DI/RO/BANK/684J741 SH 2/055.03.LP3718.004
Given the following:

- Unit 1 is at Rated Thermal Power
- 1HC-142 Charging Line Flow Controller, fails as shown below:



The expected plant response is that . . .

A. Charging flow indication on 1FI-128 and all seal injection flow indications go higher.

B. Charging flow indication on 1FI-128 and all seal injection flow indications go to zero.

C✓ Charging flow indication on 1FI-128 goes to zero and all seal injection flow indications go higher.

D. Charging flow indication on 1FI-128 goes higher and all seal injection flow indications go to zero.

RO Tier 1 Group 1

Source: Bank

Question History:
2012 Diablo Canyon NRC RO 43

K/A:
022AA1.02 Loss of Reactor Coolant Makeup
Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: **CVCS charging low flow alarm, sensor, and indicator** (Imp 3.0/2.9)

Cognitive Level:
Comprehension 2-RI. The student must recall the normal lineup of the CVCS system, and determine what the failure of the charging flow controller will have on both charging flow to the RCS, and seal injection flow to the RCPs.

Reference:
684J741 Sh 2, Chemical and Volume Control P&ID, Rev 76

Proposed reference to be provided to the applicants during examination:
None

Justification:

This valve is located on the charging injection line, if it fails closed, this will cause charging flow to lower or stop, and cause an increase to the seal injection flow rate to the RCPs.

A **INCORRECT:** Plausible This would occur if it was a charging pump speed controller which failed in this manner.

B **INCORRECT:** Plausible if the student has a misconception about the failure of this valve and the effects, since charging flow will go towards zero, the student may apply this logic to the RCP seal flow as well

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible as this is the effect if the valve fails in the opposite manner (open vice closed).

Learning Objective:

Given access to the Site specific Simulator or plant conditions, DIAGNOSE and RESPOND to CVCS malfunctions in accordance with the appropriate procedure(s).
(055.03.LP3718.004)

7. 2017 NRC 007/APE/025AA2.05/3.1/1-B/RO/MODIFIED/SEP-1/031.03.LP2189.008

Given the following:

- Unit 1 is shutdown for refueling
- Reactor Vessel level is at 22% (3/4 pipe)
- Initial RCS temp was 80°F
- 'A' RHR pump was running, became air bound and was manually secured
  - The crew responded IAW SEP-1, Degraded RHR System Capability
- 'A' RHR pump was successfully restarted
- Sweeping air from the RHR system has been completed
- The crew now wishes to return the plant to the pre-event temperature using 'A' RHR pump

Which of the following correctly identifies
(1) the RHR pump flow directed by SEP-1, Degraded RHR System Capability,
AND
(2) the MAXIMUM allowed RCS cooldown rate, while returning the plant to its initial conditions?

A✓ (1) 900-1100 GPM flow
(2) 100ºF/hr cooldown rate

B. (1) 900-1100 GPM flow
(2) 200ºF/hr cooldown rate

C. (1) 1550 GPM flow
(2) 100ºF/hr cooldown rate

D. (1) 1550 GPM flow
(2) 200ºF/hr cooldown rate

RO Tier 1 Group 1

Source: Modified

Question History:
2005 PBNP NRC RO 7

K/A:
025AA2.05 Loss of Residual Heat Removal System (RHRS)
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: **Limitations on LPI flow and temperature rates of change.** (Imp 3.1/3.5)

Cognitive Level:
Knowledge 1-B: The operator must understand the initial conditions, what the purpose and strategy of the procedure is and what the bases is for the re-establishment of RHR.

Reference:
SEP-1 Unit 1, Degraded RHR System Capability, Rev 14, Step 2,
TRM 2.2, Pressure Temperature Limits Report, Rev 10, Section 2.1.1

Proposed reference to be provided to the applicants during examination:
None

Original Question
Consider the following Unit 1 conditions:

- *Unit 1 is shutdown for refueling.*
- *Initial RCS temp was 80ºF.*
- *'A' RHR pump was running but became air bound, crew responded IAW SEP-1, Degraded RHR System Capability.*
- *'A' RHR pump was successfully restarted after about 30 minutes.*
- *Final RCS temp is 110ºF.*

The crew now wishes to return the plant to the pre-event temperature using 'A' RHR pump.

*Which of the following correctly identifies the RHR pump flow directed by SEP-1 and **MAXIMUM** allowed RCS cooldown rate while returning the plant to its initial conditions?*

A) 450-650 GPM flow; 50ºF/hr cooldown rate

B) 900-1100 GPM flow; 100ºF/hr cooldown rate

C) 1400-1600 GPM flow; 200ºF/hr cooldown rate

D) 2800-3000 GPM flow; Maximum cooldown rate does not apply while in SEP-1.

Proposed Answer: B.

Justification:

The flow rate is dictated by SEP-1 while the cooldown rate is driven by TS, nothing in SEP-1 dictates a cooldown rate.

A **CORRECT:** See above explanation.

B **INCORRECT:** The flow rate is correct, but the cooldown rate is less restrictive than procedurally directed. Plausible because the cooldown rate given is the one given for pressurizer cooldown limits.

C **INCORRECT:** The flow is incorrect, and the cooldown rate is the correct rate. Plausible because the flow is called out to sweep the system of air and the cooldown rate is correct.

D **INCORRECT:** The flow rate and cooldown rate are incorrect. Plausible because the flow rate is called out to sweep the system of air and the cooldown rate given is the one given for pressurizer cooldown limits

Learning Objective:

Given a specific set of plant parameters or access to the Site Specific Simulator and appropriate plant/system condition, IMPLEMENT the following procedures:
a. SEP-1
b. SEP-1.1
(031.03.LP2189.008)

8. 2017 NRC 008/APE/027AG2.2.12/3.7/3-SPR/RO/NEW/TS 3.4.9/051.01.LP0078.009

Given the following:

- Unit 1 is in MODE 1
- Pressurizer Heater Group ‘D’ is out of service
- 1T-1C, Pressurizer Heater Group Energy Input Test Unit 1 is being performed on the Group 'C' heater group

Given the reference and data provided

(1) What is the calculated pressurizer heater energy entered in the Acceptance Criteria section,

AND

(2) Only taking Pressurizer Heater Group ‘C’ into consideration, what is the status of Technical Specification 3.4.9: Pressurizer?

A. 99 KW MET

B. 106 KW MET

C.✓ 99 KW NOT MET

D. 106 KW NOT MET

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
027AG2.2.12: Pressurizer Pressure Control System (PZR PCS) Malfunction
Knowledge of surveillance procedures. (Imp 3.7/4.1)

Cognitive Level:
Comprehension 3-SPR: The operator must correctly calculate the pressurizer heater power given the reading from the surveillance test (TS 43), recall the required value of pressurizer heater power from TS 3.4.9 and analyze the calculated result against the TS requirement.

Reference:

TS 43 T-1C Unit 1, 1T-1C Pressurizer Heater Group Energy Input Test, Rev 3, Steps 5.1.17, and 6.1
TS 3.4.9, Pressurizer, Rev 3, LCO b.

Proposed reference to be provided to the applicants during examination:

Provide page 7 of TS-43 with the first 4 steps (5.1.13-51.16), and data for the fifth step (5.1.16) filled in, but NOT TOTALED UP

Justification:

99 KW is the correct calculated value, and this would not meet the requirements of the TS. TS 3.4.9 requires pressurizer heater power to be $\geq$ 100KW

A **INCORRECT:** The calculated value is correct, but the TS is not met. Plausible if the student incorrectly recalls the TS requirements.

B **INCORRECT:** The calculated value and TS status are incorrect. Plausible if the student does not apply the uncertainty factor as required by step 5.1.17, and this value would meet the TS.

C **CORRECT:** See above explanation.

D **INCORRECT:** The calculated value is incorrect and TS status is correct. Plausible if the student correctly applies the uncertainty factor and incorrectly recalls the TS requirements.

Learning Objective:

Identify and discuss the technical specifications associated with pressurizer, level control, pressure control, and relief system components, parameters, and operations to include:
a. LCO
b. LCO Applicability
(051.01.LP0078.009)

9. 2017 NRC 009/EPE/038EK1.01/3.1/3-SPR/RO/BANK/STEAM TABLES/031.02.LP0441.016
Given the following:

- A Steam Generator Tube Rupture (SGTR) has occurred
- EOP-3, Steam Generator Tube Rupture is being performed
- Reactor Coolant pumps are running
- The crew has isolated the ruptured Steam Generator and has completed the cooldown to a target temperature of 475°F
- The OS directs you to stabilize Reactor Coolant System (RCS) temperature and set up the Condenser Steam Dump Controller to maintain RCS temperature at approximately 470°F

Which of the following is the correct Condenser Steam Dump Controller setpoint required to maintain RCS temperature at approximately 470°F?

A.✓ 500 psig

B. 514 psig

C. 530 psig

D. 540 psig

RO Tier 1 Group 1

Source: Bank

Question History:
2012 DC Cook NRC RO 48

K/A:
038EK1.01 Steam Generator Tube Rupture
Knowledge of the operational implications of the following concepts as they apply to the SGTR: **Use of steam tables.** (Imp 3.1/3.4)

Cognitive Level:
Comprehension 3-SPR: The student must determine the saturation setting for the steam dumps utilizing steam tables.

Reference:
Steam Tables

Proposed reference to be provided to the applicants during examination:
Steam Tables

Justification:

514.56 psia is Psat for 470°F. The Controller would need to be set at 500 psig. (514.56 -14.7 = 499.85 or 500 given controller readability).

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible because 514.56 psia is Psat for 470°F.

C **INCORRECT:** Plausible because 530 psig is derived from adding 14.7 psi to 515 psia.

D **INCORRECT:** Plausible because 540 psia is Psat for 475°F (Absolute not PSIG).

Learning Objective:

Given access to Site Specific Simulator, IMPLEMENT the EOPs to respond to a faulted ruptured Steam Generator.
(031.02.LP0441.016)

10. 2017 NRC 010/APE/040AK2.01/2.6/1-I/RO/NEW/008D195 SH 7/052.02.LP0153.003

Given the following:

- A Unit 2 reactor trip and safety injection occurred from 7% reactor power
- Containment pressure is 13 psig and RISING SLOWLY
- 2C04 2A 4-1, REACTOR COOLANT TAVG LOW annunciator is **LIT**
- 2P-1A and 2P-1B, Reactor Coolant pumps are running
- Pressurizer level is 0%
- ‘A’ SG Narrow Range is 10% and LOWERING
- ‘B’ SG Narrow Range is 30% and RISING (lowest value was 28%)
- ‘A’ Steam Pressure is 750 psig and LOWERING
- ‘B’ Steam Pressure is 915 psig and LOWERING
- ‘A’ Steam Flow is 0.9×10^6 lbm/hr
- ‘B’ Steam Flow is 0 lbm/hr
- Safety Injection has **not** been reset

Which of the following would be expected to have AUTOMATICALLY occurred?

A.✓ ‘A’ MSIV only shut

B. ‘B’ MSIV only shut

C. Both MSIVs shut

D. Neither MSIV shut

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
040AK2.01 Steam Line Rupture
Knowledge of the interrelations between the Steam Line Rupture and the following: **Valves.** (Imp 2.6/2.5)

Cognitive Level:
Knowledge 1-I: The student must recall the interlocks which will cause the MSIV (respective or both) to automatically close given the initial condition.

Reference:
883D195 Sh 7, Safeguards Actuation Signals Logic Diagram, Rev 25
883D195 Sh 15, RC Trip Signal Logic Diagram, Rev 9
STPT 2.2, Steam Line Isolation Setpoint Document, Rev 7, Section 1.0 and 1.2
ARB 2C04 2A 4-1, REACTOR COOLANT TAVG LOW, Rev 5, Section 2

Proposed reference to be provided to the applicants during examination:
None

Justification:
The following signals will cause a closure of the “respective” MSIV
SI with Hi Steam Flow ($0.52x10^6$ lbm/hr) and TAVG <543°F
SI with HI HI Steam Flow ($4.85x10^6$lbm/hr)

The following signal will cause the closure of both MSIVs
15 psig Containment Pressure

A **CORRECT:** See explanation above.

B **INCORRECT:** Plausible if the student determines that the initial conditions show that the ‘B’ MSIV has shut, and that is the action which automatically occurred.

C **INCORRECT:** Plausible if the student incorrectly recalls the containment pressure setpoint determining both should have shut.

D **INCORRECT:** Plausible if the student incorrectly recalls the Low TAVE annunciator value.

Learning Objective:
DESCRIBE the interlocks associated with the Main Steam System and components.
(052.02.LP0153.003)

11. 2017 NRC 011/APE/056AK3.02/4.4/2-RI/RO/BANK/EOP-3.1/031.02.LP0441.002
Given the following:

- Unit 2 operators are responding to a Steam Generator Tube Rupture in the ‘A’ Steam Generator concurrent with a loss of offsite power (LOOP)
- The crew transitioned to EOP-3.1, Post-Steam Generator Tube Rupture Cooldown Using Backfill and commenced a cooldown to cold shutdown
- Operators are currently refilling the ‘A’ SG to 73% when offsite power is restored
- The Operating Supervisor suggests starting the 'A' reactor coolant pump (RCP) to enhance the recovery process
- The 'A' RCP would be the first RCP started

Which of the following would be the correct response and basis for the existing plant conditions?

A. Agree with starting the ‘A’ RCP.
RCP ‘A’ is the preferred RCP to provide normal pressurizer spray flow.

B. Agree with starting the ‘A’ RCP.
RCP operation is preferred during backfill to ensure homogeneous fluid temperatures.

C. Disagree with starting the ‘A’ RCP.
RCP operation may result in RCP seal damage.

D.✓ Disagree with starting the ‘A’ RCP.
Starting RCP 'A' may result in an inadvertent criticality.

RO Tier 1 Group 1

Source: Bank

Question History:
2011 PBNP NRC RO 14

K/A:
056AK3.02 Loss of Offsite Power
Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: **Actions contained in EOP for loss of offsite power.**
(Imp 4.4/4.7)

Cognitive Level:
Comprehension 2-RI: The operator must understand the initial conditions, what effect the LOOP has had, where the crew is within the EOP network, and what the mitigating strategies are for EOP-3.1.

Reference:
EOP-3.1 Unit 2, Post-Steam Generator Tube Rupture Cooldown Using Backfill Rev 29, Step 1 Caution-1, and Step 1
BG-EOP-3.1, Post-Steam Generator Tube Rupture Cooldown Using Backfill, Rev 22, Step 1-Caution 1, and Step 1

Proposed reference to be provided to the applicants during examination:
None

Justification:

The first step of the EOP is to establish forced circulation, or verify natural circulation flow if RCPs cannot be started. With the loss of off-site power, both RCPs have stopped, so starting one will aid in the mitigation of the event. The step is also modified by several cautions and notes. One caution states inadvertent criticality may occur following any natural circulation cooldown if the first RCP started is in the ruptured loop.

A **INCORRECT:** Wrong recommendation and reason. Plausible if the student has a misconception on which pump to start, the 'B' RCP is the preferred RCP for normal pressurizer spray flow.

B **INCORRECT:** Wrong recommendation and reason. Plausible if the student has a misconception on which pump to start, this is the correct reason a RCP should be started on the ‘A’ loop.

C **INCORRECT:** Correct recommendation and wrong reason. Plausible if the student has a misconception that a LOOP would have caused a loss of cooling to the RCP when none was stated in the question stem.

D **CORRECT:** See above explanation.

Learning Objective:

LIST the major actions accomplished by the following procedures:

a. EOP 2
b. EOP 3
c. EOP 3.1
d. EOP 3.2
e. EOP 3.3

(031.02.LP0441.002)

12. 2017 NRC 012/APE/058AA1.01/3.4/1-S/RO/BANK/AOP-0.0/054.03.LP0121.010
Given the following:

- During a maintenance period the following DC system alignment exists:
  - DC Bus D02 is aligned to D09, Swing Battery Charger and D305 Swing Battery
- D09, Swing Battery Charger develops a ground and the AC power breaker on D09 Trips

Which of the following methods can be used to restore D02 to an acceptable alignment?

A. Crosstie DC Bus D02 to DC Bus D04

B. Align 2D-205, Battery and 2D-207, Battery Charger to D-02

C.✓ Restore D08, Battery Charger and D06, Battery to DC Bus D02

D. Align Swing Battery Charger D109 to DC Bus D02 and D305 Battery

RO Tier 1 Group 1

Source: Bank

Question History:
2015 PBNP NRC Exam RO 12 Past 2 NRC Exams

K/A:
058AA1.01 Loss of DC Power
Ability to operate and / or monitor the following as they apply to the Loss of DC Power: **Cross-tie of the affected dc bus with the alternate supply.** (Imp 3.4/3.5)

Cognitive Level:
Knowledge 1-S: The operator recall the DC system, understand the current DC system line up and the effects that the trip will have on the system.

Reference:
AOP-0.0, Vital DC System Malfunction, Rev 35, Step 5
0-SOP-DC-002, 125 VDC System Bus D-02 & Components, Rev 23, Section 5.5

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions, this is the method called out in the Abnormal Operating Procedure, and is achievable based on system design and interlocks.

A **INCORRECT:** System interlocks will prevent this lineup. Plausible if the student does not understand the interlock and interconnections of D02 and D04.

B **INCORRECT:** System interlocks will prevent this lineup. Plausible if the student does not understand how the system Kirk Key interlocks function.

C **CORRECT:** See above explanation.

D **INCORRECT:** System interlocks will prevent this lineup. Plausible if the student does not understand the interlock and connections of D301A.

Learning Objective:

STATE why train cross-connection is not allowed.
(054.03.LP0121.010)

13. 2017 NRC 013/APE/062AA2.04/2.5/1-I/RO/NEW/1-SOP-CC-001/051.06.LP0084.006
Given the following:

- Unit 1 is operating at Rated Thermal Power
- Blockage is developing in the service water line supplying the component cooling heat exchanger aligned to Unit 1
- Procedure 1-SOP-CC-001, Component Cooling System requires service water be adjusted to the CC Heat Exchangers to control the CC supply header temperature.

Times must be logged if the CC supply header temperature exceeds the maximum normal temperature of ___(1)___. In the case of short-term additional heat load from RHR, the CC supply header temperature to the RCPs can exceed the normal limit for no more than two hours and shall not exceed ___(2)___.

	(1) Maximum Normal Limit	(2) Two-Hour Short Term Limit
A.✓	105°F	125°F
B.	105°F	145°F
C.	125°F	145°F
D.	125°F	175°F

RO Tier 1 Group 1

Source: New

Question History:
None

K/A:
062AA2.04 Loss of Nuclear Service Water
Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: **The normal values and upper limits for the temperatures of the components cooled by SWS.** (Imp 2.5/2.9)

Cognitive Level:
Knowledge 1-I: The student must recall the procedural limits for Component Cooling header temperature, as controlled by service water to the CC heat exchanger.

Reference:
AOP-9A Service Water System Malfunction, Rev. 34, Step 19
1-SOP-CC-001 Component Cooling System, Rev 29, Section 3.13, and 3.20

Proposed reference to be provided to the applicants during examination:
None

Justification:
The Abnormal Operating Procedure for loss of service water directs use of operating instructions to address adverse temperature trends for service water cooled equipment. The component cooling system operating procedure provides a normal CC supply header control band of 90°F to 105°F. In cases of increased heat load (e.g., RHR system startup), the CC supply temperature to the RCPs is allowed to reach 125°F for no more than two hours.

A **CORRECT:** See above explanation.

B **INCORRECT:** The first half is correct. the second half is plausible if the student mistakenly applies the CVCS letdown diversion temperature as the extended limit (SW cools CC, and CC cools letdown).

C **INCORRECT:** Plausible that the student only recalls the 125°F limit as the upper band. The 145°F limit is plausible if the student mistakenly applies the CVCS letdown diversion temperature as the extended limit (SW cools CC, and CC cools letdown).

D **CORRECT:** Plausible that the student only recalls the 125°F limit as the upper band. The 175°F limit is plausible if the student mistakenly applies the RHR HX CC outlet temperature limit to this question. (SW cools CC, and CC cools RHR).

Learning Objective:
RECOGNIZE and ANALYZE the response of the Component Cooling Water System to a loss of the Service Water system.
(051.06.LP0084.006)

14. 2017 NRC 014/APE/065AA2.05/3.4/1-P/RO/BANK/AOP-5B/055.03.LP2439.005
Given the following:

- Unit 1 is at Rated Thermal Power
- Annunciator C01A 1-9, INSTRUMENT AIR HEADER PRESSURE LOW is **LIT**

If a ruptured Instrument Air header is causing a continuous lowering of Instrument Air header pressure, which of the following will require a Reactor Trip per AOP-5B, Loss of Instrument Air?

A. Loss of Feedwater Heater Level control

B. Loss of Pressurizer Spray valve control

C. Loss of Letdown Orifice Isolation valve control

D.✓ Loss of Main Feedwater flow control valves control

RO Tier 1 Group 1

Source: Bank

Question History:
2005 PBNP NRC RO 16

K/A:
065AA2.05 Loss of Instrument Air
Ability to determine and interpret the following as they apply to the Loss of Instrument Air: **When to commence plant shutdown if instrument air pressure is decreasing.** (Imp 3.4/4.1)

Cognitive Level:
Knowledge 1-P: The operator must understand the initial conditions of the event, and apply the mitigation strategy of the AOP.

Reference:
AOP-5B, Loss of Instrument Air, Rev 46, Step 5

Proposed reference to be provided to the applicants during examination:
None

Justification:

AOP-5B directs the crew to trip the reactor (s) if 'Main feedwater flow control valves operating at required' is not met.

A **INCORRECT:** Plausible, as this may cause a feedwater system induced transient on the plant, but is does not require a reactor trip.

B **INCORRECT:** Plausible as a loss of air would cause the spray valves to fail in the closed position, but that would not require the crew to trip the reactor.

C **INCORRECT:** Plausible as a loss if air would cause the orifice isolation valves to fail shut, but that wouldn't require a reactor trip.

D **CORRECT:** See above explanation.

Learning Objective:

Given access to the Site Specific Simulator or specific plant conditions, APPLY the appropriate guidance provided in the applicable AOPs for various system/component malfunctions.
(055.03.LP2439.005)

15. 2017 NRC 015/EPE/E04G2.1.31/4.6/2-RI/RO/BANK/EOP-1.4/031.02.LP0465.008
Given the following:

- Unit 1 was operating at Rated Thermal Power
- A Loss of Coolant Accident has occurred
- Actions of EOP-1.4, Transfer to Containment Sump Recirculation – High Head Injection have been completed
- Both RHR and SI Pumps are running and both trains are on recirc
- RWST level is 28%
- Annunciator 1C20A 4-4, UNIT 1 OR 2 RHR PUMP ROOMS LEVEL HIGH **LIT**
- Amber light for 1WL-4100, 1P-10B RHR Pump Drain To Sump, is **LIT**
- Amber light for 1WL-4101, 1P-10A RHR Pump Drain To Sump, is **NOT** LIT
- The crew placed 1WL-4100 control switch in the OPEN position
- 1WL-4101 control switch is in the CLOSE position
- P-40A, -19' Sump Pump, is running continuously

Based on the above indications, the crew is required to?

(1P-10B, Residual Heat Removal Pump)
(1P-15B, Safety Injection Pump)

A. Stop all Safety Injection and RHR pumps.

B.✓ Stop 1P-10B and 1P-15B.

C. Stop 1P-10B only.

D. Continue operating with all Safety Injection and RHR pumps.

RO Tier 1 Group 1

Source: Bank

Question History:
2003 PBNP NRC RO 16

K/A:
E04G2.1.31 LOCA Outside Containment
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (Imp 4.6/4.3)

Cognitive Level:
Comprehension 2-RI: The operator must understand the initial conditions, determine the meaning of the amber lights and running pump, then determine the action required to mitigate the situation.

Reference:
EOP-1.4 Unit 1, Transfer to Containment Sump Recirculation – High Head Injection, Rev 31, Step 37
BG-EOP-1.4, Background Document Transfer to Containment Sump Recirculation – High Head Injection, Rev 19, EOP Step 37

Proposed reference to be provided to the applicants during examination:
None

Justification:
Given the amber light on and the running sump pump, the indications are that 'B' RHR seal has failed, based on that decision, the RHR pump needs to be secured. The procedure directs securing the "train" of pumps if an RHR pump leakage is suspected.

A **INCORRECT:** Plausible if the student has a misconception of the purpose of the procedure, since the procedure isolates flow paths while attempting to diagnose the leak, and since this is a LOCA outside containment, the break flow should be lower based on pipe size.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible as this is the only pump with suspected leakage and the student may fail to apply knowledge of the system alignment on high head recirculation.

D **INCORRECT:** Plausible if the student has the misconception that with the pump running continuously, the pump is maintaining the leakage at an acceptable level and pump use may continue.

Learning Objective:
Given appropriate conditions/parameters and access to the site specific Simulator, IMPLEMENT the following procedures for the specified conditions:
a. ECA-1.1 to respond to a loss of Containment Sump Recirculation
b. ECA-1.2 to respond to an intersystem LOCA
c. ECA-1.3 to respond to containment sump blockage
d. ECA-2.1 to respond to both Steam Generators being faulted
(031.02.LP0465.008)

16. 2017 NRC 016/EPE/E11G2.4.46/4.2/3-SPK/RO/BANK/ECA-1.1/031.02.LP0465.008
Given the following:

- Unit 1 has had a Design Basis LOCA coincident with a loss of 1X04, Low Voltage Station Auxiliary Transformer
- Plant conditions are as follows:
  - Containment pressure is 6 psig
  - While establishing sump recirc alignment, 1SI-850A, Cont Sump B Isolation Valve was NOT able to be opened
- The crew is implementing EOP-1, Loss of Reactor or Secondary Coolant, while waiting on RWST level which will allow establishing recirc flow

- The following annunciators have just been received:
  - C02 E 1-2, G-03 EMERGENCY DIESEL TRIP OR LOCKOUT
  - C02 E 2-2, G-03 EMERGENCY DIESEL
  - C01 B 2-9, 1T-13 RWST LEVEL LOW

Which of the following is:
(1) the correct status of the emergency recirculation capability?
AND
(2) the action(s) the applicable procedure(s) will direct?

A. (1) Only ONE train of emergency recirculation capability has been lost
(2) Align one train of RHR AND Containment Spray for recirculation

B. (1) Only ONE train of emergency recirculation capability has been lost
(2) Align one train of RHR for sump recirculation but do NOT align Containment Spray for recirculation at this time

C.✓ (1) BOTH trains of emergency recirculation capability have been lost
(2) Secure Containment Spray AND establish minimum injection flow required to remove decay heat

D. (1) BOTH trains of emergency recirculation capability have been lost
(2) Secure Containment Spray, RHR, and Safety Injection pumps AND establish maximum charging

RO Tier 1 Group 1

Source: Bank

Question History:
2010 Farley NRC RO 73

K/A:
E11G2.4.46 Loss of Emergency Coolant Recirculation
Ability to verify that the alarms are consistent with the plant conditions.
(Imp 4.2/4.2)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, determine what the various alarms mean, determine what plant conditions would cause those alarms was that mitigating strategy would be employed in the EOP network given those conditions.

Reference:
EOP-1.3 Unit 1, Transfer to Containment Sump Recirculation, Rev 55, Foldout Page Item 2
ECA-1.1 Unit 1, Loss of Containment Sump Recirculation, Rev 41, Steps 3 and 13, and Foldout Page Item 2
STPT 11.1, Safety Injection System General Instrumentation Channels Setpoint Document, Rev 20, Channel 1CL-972L/973L

Proposed reference to be provided to the applicants during examination:
None

Justification:

Based on the initial conditions, one train has been aligned for recirc, and the crew is currently waiting for the RWST to be <17%, to align spray for recirc also. With the inability to open 1SI-850A, and the loss of the EDG, both trains have lost the capability to be lined up for recirc. A transition to ECA-1.1 is necessary, and the spray pumps will be secured, and charging will be adjusted to attempt to get the required amount of flow to remove decay heat.

A **INCORRECT:** Both parts are incorrect. Plausible if the student does not understand both trains of recirc have been lost. The second part of the answer would be correct if the first part was true, if only one train had been lost, then these are the correct actions.

B **INCORRECT:** Both parts are incorrect. Plausible if the student does not understand both trains of recirc have been lost and the student misdiagnosed the tank level based on alarm indications because the second part of the answer would be true when the RWST level is >17%.

C **CORRECT:** See above explanation.

D **INCORRECT:** The first part is correct, and second part is incorrect. Plausible if the student misdiagnosed the tank level based on alarm indications because this would be the correct course of action if the RWST is <10%.

Learning Objective:

Given appropriate conditions/parameters or access to the site specific Simulator, IMPLEMENT the following procedures for the specified conditions:

a. ECA-1.1 to respond to a loss of Containment Sump Recirculation
b. ECA-1.2 to respond to an intersystem LOCA
c. ECA-1.3 to respond to containment sump blockage
d. ECA-2.1 to respond to both Steam Generators being faulted

(031.02.LP0465.008)

17. 2017 NRC 017/EPE/E05EK1.01/3.8/1-B/RO/BANK/BG-CSP-H.1/043.03.LP1998.003

Given the following:

- A loss of Feedwater has resulted in a reactor trip
- The crew is performing actions of CSP-H.1, Response to Loss of Secondary Heat Sink.
- Reactor Coolant pumps have been stopped

Which of the following describes the operation of the PRZR PORVs during this event?

A. Remain closed with block valves closed to preserve RCS inventory unless conditions exist that require initiation of Bleed and Feed cooling of the RCS.

B. One PORV is opened if Bleed and Feed cooling is required, with the other PORV and block valve closed and remaining available for automatic control.

C.✓ May be operated to depressurize the RCS so that Low Steam Pressure and Low PRZR Pressure SI signals can be blocked; this prevents unwanted SI actuation when depressurizing SGs prior to performing action to establish Condensate flow.

D. May be operated to depressurize the RCS so that Low Steam Pressure and Low PRZR Pressure SI signals can be blocked; this prevents unwanted SI actuation when depressurizing SGs prior to performing action to establish Main Feedwater flow.

RO Tier 1 Group 1

Source: Bank

Question History:
2010 Ginna NRC RO 41

K/A:
E05EK1.01 Loss of Secondary Heat Sink
Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink)
Components, capacity, and function of emergency systems. (Imp 3.8/4.1)

Cognitive Level:
Knowledge 1-B: The student must recall the mitigation strategy of the CSP when aligning feed and how the PORVs will be utilized during the procedure.

Reference:

BG-CSP-H.1, Background for Response to Loss of Secondary Heat Sink, Rev 29, EOP Step 22 and 23

CSP-H.1 Unit 1, Response to Loss of Secondary Heat Sink, Rev 42, Step 22 and 23

Proposed reference to be provided to the applicants during examination:

None

Justification:

Aux Spray is the preferred method, but one PORV may be used if Aux Spray is not available.

A **INCORRECT:** Plausible because opening a PORV can possibly lead to a loss of inventory. The only way to block the valves will be to close a block valve. The only time block valves are procedurally closed is for a leaking or failed PORV.

B **NCORRECT:** Plausible since this the use of one PORV is procedurally allowed, but it is not the reason the other PORV remains closed.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible because the phrasing is similar, but no depressurization is required in order to initiate Feedwater flow.

Learning Objective:

STATE how RCS Bleed and Feed is initiated and how it provides adequate core cooling.
(043.03.LP1998.003)

18. 2017 NRC 018/APE/077AK1.02/3.3/3-SPR/RO/BANK/OP 2A/055.01.LP0258.002
Given the following:

- Unit 2 is in MODE 1
- Grid disturbances have required the crew to lower turbine generator output to 470 MWE
- Generator Hydrogen pressure is 60 psig

Which of the following is the maximum VAR loading per the capability curve the generator may carry without exceeding limits?
(See provided reference)

A. 235 Mvars overexcited

B.✓ 295 Mvars overexcited

C. 320 Mvars overexcited

D. 330 Mvars overexcited

RO Tier 1 Group 1

Source: Bank

Question History:
2015 PBNP NRC Exam RO 18 Past 2 NRC Exams

K/A:
077AK1.02 Generator Voltage and Electric Grid Disturbances
Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: **Over-excitation.**
(Imp 3.3/3.4)

Cognitive Level:
Comprehension 3-SPR: The operator takes the initial conditions, uses the reference material and determines the correct limits for the generator.

Reference:
OP 2A Unit 2, Normal Power Operations, Rev 11, Figure 3

Proposed reference to be provided to the applicants during examination:
OP 2A Unit 2, Figure 3, page 64

Justification:

300 Mvars overexcited is the result of using Figure 3 with the intersection being at 470 MWE and 60 psig hydrogen, so a value of 295 was selected due to graph readability.

A **INCORRECT:** 235 Mvars overexcited is the result of using Figure 3 with the intersection being at 470 MWE and 45 psig hydrogen. Plausible if the student uses the wrong pressure curve.

B **CORRECT:** See above explanation.

C **INCORRECT:** 320 Mvars overexcited is the result of using Figure 3 with the intersection at 470 MWE and 75 psig hydrogen, but using the lower or underexcited portion of the graph. Plausible if the student uses the wrong half of Figure 3.

D **INCORRECT:** 330 Mvars overexcited is the result of using Figure 3 with the intersection being at 470 MWE and 75 psig hydrogen, or the normal curve on the graph. Plausible if the student has a misconception of how to use the Figure 3.

Learning Objective:

Given appropriate system/equipment conditions and indications, DESCRIBE and DEMONSTRATE the ability to perform/control the following evolutions:
a. Increase power as Xenon changes
b. Control Delta Flux
c. Determine Reactor Thermal output
d. Control MVARS as directed by the System Control Supervisor
e. Predict Xenon transients
f. Estimate Rod position at power and during power swings such that Delta Flux remains within Technical Specifications limits
g. Transfer 4 kV non-vital busses between transformers
(055.01.LP0258.002)

19. 2017 NRC 019/APE/003AG2.1.27/3.9/2-DR/RO/BANK/PBN LP0576/035.01.LP0576.007
Given the following:

- Unit 1 is conducting a mid-cycle reactor startup
- Shutdown Bank A and B rods are at 228 steps (full out)
- Control Bank A rods are at 228 steps (full out)
- Control Bank B rods are at 228 steps (full out)
- Control Bank C rods are at 155 steps
- Control Bank D rods are at 30 steps
- Control Bank D rod C7 drops into the core

Will annunciator 1C04 1A 1-5, ROD BOTTOM ROD DROP alarm and why?

A. No, the annunciator is automatically bypassed when reactor power is below the permissive P-6 setpoint

B✓ No, the annunciator is automatically bypassed when the Bank D P/A Converter is below 35 steps

C. Yes, the annunciator is activated once Control Bank A rods are greater than 35 steps from the bottom

D. Yes, the IRPI rod bottom lights are one of the causes for the annunciator

RO Tier 1 Group 2

Source: Bank

Question History:
None

K/A:
003AG2.1.27 Dropped Control Rod
Knowledge of system purpose and/or function. (Imp 3.9/4.0)

Cognitive Level:
Comprehension 2-DR: The student must understand the initial conditions, determine the impact of the dropped rod, and if it will produce the specified annunciator.

Reference:
PBN LP0576, Rod Position Indication System, Rev 15, Section II.B.7

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the conditions of the IRPI being greater than 20 steps from the bottom, prior to the dropped rod, and that the P/A converter is not greater than 35, and one of the rods (C7) was dropped, then no the annunciator would not be lit.

A **INCORRECT:** Correct indication and wrong reason. Plausible if the student has a misconception of what items are defeated by the P-6 permissive.

B **CORRECT:** See above explanation.

C **INCORRECT:** Wrong indication and reason. Plausible if the student has the misconception that the annunciator is activated for all control banks when the rods reach 35 steps on control bank A

D **INCORRECT:** Wrong indication and reason. Plausible if the student has a misconception of what indication feed that annunciator, as this is one of the times that the rod bottom light will extinguish, then re-energize but the rod dropped annunciator will not come in.

Learning Objective:

DESCRIBE system response to the following:

a. Rod Bottom Bistable Trip
b. Inoperable, Misaligned, or Dropped Rod
c. Rod Position Indication component failures
d. Loss of Control Rod Drive Mechanism cooling

(053.01.LP0576.007)

20. 2017 NRC 020/APE/024AK1.01/3.4/2-DR/RO/BANK/NUC GFP RXT 008/NUC GFP RXT 008.021

Given the following:

- A component failure during testing resulted in an automatic trip signal on High Nuclear Power
- The reactor failed to trip and the crew has entered CSP-S.1, Response To Nuclear Power Generation/ATWS
- The Main Turbine is tripped and both AFW Pumps are running
- Emergency Boration has been initiated
- TAVG is 590°F

Assuming NO automatic or manual control rod motion, which of the following describes the trend of TAVG due to plant response and the RCS boration?

TAVG will . . .

A. lower continuously until Emergency Boration is stopped.

B. remain the same until reactor power is less than AFW heat removal capability and then begin to lower.

C. remain the same until control rods drop, then lower rapidly and stabilize when steam dumps are closed.

D.✓ lower continuously until steam dumps gain control of TAVG and then stabilize as steam dumps modulate closed.

RO Tier 1 Group 2

Source: Bank

Question History:

2010 Ginna NRC RO 57

K/A:

024AK1.01 Emergency Boration
Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: **Relationship between boron addition and change in T-ave.** (Imp 3.4/3.8)

Cognitive Level:

Comprehension 2-DR: The student must understand the initial conditions, determine the effect of the boration, and determine what the effect will be on TAVE given no rod motion or additional operator action.

Reference:

NUC GFP RXT 008 Reactor Operational Physics, Rev 0, Section 4.3
WCAP 8330 Anticipated Transients Without Trip Analysis, Appendix B

Proposed reference to be provided to the applicants during examination:

None

Justification:

When the Emergency Boration reaches the reactor the effect is similar to but not as rapid as the negative reactivity insertion of control rod motion. With the steam dumps wide open, TAVG will lower until they gain control. At that point the steam dumps will modulate closed as TAVG is attempting to lower due to boration. With the steam dumps modulating closed to control TAVG at no-load temperature, TAVG will stabilize with dumps finally open only as needed to remove decay heat and reactor coolant pump heat.

A **INCORRECT:** Plausible because TAVG will begin lowering as the Emergency Boration reaches the reactor. However, it has no effect on TAVG once the reactor is sub-critical.

B **INCORRECT:** Plausible from a heat balance standpoint, if the student has the conception that only power is changing. However, TAVG will be lowering continuously from the negative reactivity insertion.

C **INCORRECT:** Plausible because this is an instantaneous response on a normal plant trip. However, TAVG will be lowering continuously from the negative reactivity insertion.

D **CORRECT:** See above explanation.

Learning Objective:

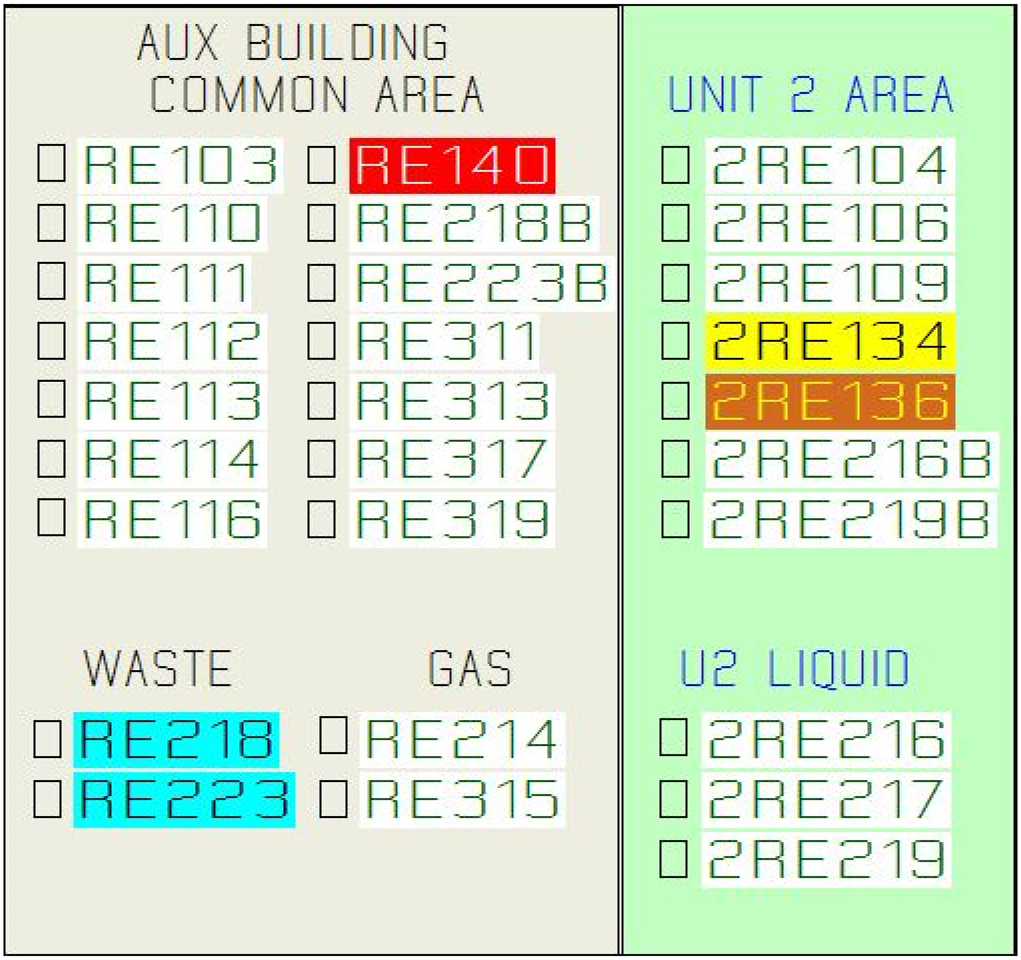
Explain the monitoring and control of Tave, Tref and power during power operation.
(NUC GFP RXT 008.020)

Explain the effects of control rod motion, boration and dilution on reactor power.
(NUC GFP RXT 008.021)

21. 2017 NRC 021/APE/61AA2.01/3.5/1-F/RO/NEW/PPCS DISPLAY/053.05.LP0286.010

Given the following:

- While monitoring the control boards, the following is noted on the Radiation Monitoring System PPCS Grid display:



Which of the following detectors is displaying the NO DATA status??

A. RE140

B. RE218

C. 2RE134

D.✓ 2RE136

RO Tier 1 Group 2

Source: New

Question History:
None

K/A:
061AA2.01 Area Radiation Monitoring (ARM) System Alarms
Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: **ARM panel displays.**
(Imp 3.5/3.7)

Cognitive Level:
Knowledge 1-F: The student must recall which of the detectors is showing the requested status.

Reference:
PPCS Display Legend

Proposed reference to be provided to the applicants during examination:
None

Justification:
As shown in the legend, the NO DATA status is a mustard colored background with the detector nomenclature in a yellow font.

A **INCORRECT:** Plausible the status of this monitor is High Alarm.

B **INCORRECT:** Plausible the status of this monitor is Fail.

C **INCORRECT:** Plausible the status of this monitor is Alert.

D **CORRECT:** See above explanation.

Learning Objective:
IDENTIFY the status of a particular field unit and DETERMINE the status of any channel at the System Server and the PPCS RMS Grid Screen.
(053.05.LP0286.010)

22. 2017 NRC 022/APE/68AG2.4.18/3.3/1-B/RO/NEW/BG AOP-10/055.03.LP1275.004
Given the following:

- Evacuation of the control room per AOP-10, Control Room Inaccessibility is required due to a bomb threat, NOT a fire.

Which of the following is an action taken PRIOR to control room evacuation AND the basis for that action?
(Assume all procedure steps have been performed in order)

A.✓ Align Charging Pumps to the RWST to ensure a supply of make-up.

B. Ensure one Condensate Pump is running per unit to provide adequate suction pressure for the operating Main Feed Pump.

C. Place 1(2)P-53 Motor Driven AFW Pumps in PULL-OUT, to prevent spurious operation of the pumps before local alignments can be made. They will be started and used to control steam generator level locally.

D. Adjust Atmospheric Steam Dump controllers to 1005 psig in order to assist the Condenser Steam Dumps in maintaining RCS temperature at no-load TAVE should condenser vacuum unexpectedly become degraded.

RO Tier 1 Group 2

Source: New

Question History:
None

K/A:
068AG2.4.18 Control Room Evacuation
Knowledge of the specific bases for EOPs. (Imp 3.3/4.0)

Cognitive Level:
Knowledge 1-B: The student must recall basis of AOP-10 and match the correct basis to an step from the procedure action.

Reference:
AOP-10, Control Room Inaccessibility, Rev 7, Step 5
BG AOP-10, Background Control Room Inaccessibility, Rev 3, Step 5

Proposed reference to be provided to the applicants during examination:
None

Justification:

The overall bases of AOP-10, Control Room Inaccessibility is that actions will be taken to place both units in Hot Shutdown with no other accident in progress. Some of the actions are preferred to be taken prior to leaving the control room.

A **CORRECT:** Per the AOP-10 Background, Step 5 aligns Charging Pump suctions to the RWST to ensure a virtually unlimited supply of make-up.

B **INCORRECT:** Plausible because Step 10 ensures that only one Condensate Pump is running. Per the AOP-10 Background, this is done in order to cooldown the secondary and prevent overpressurizing the condenser.

C **INCORRECT:** Plausible because Step 7 places both P-29 Motor Driven AFW Pumps in PULLOUT. Per the AOP-10 Background, this is done because there is no local S/G level indication for the motor driven AFW pumps.

D **INCORRECT:** Plausible because Step 4 adjusts Atmospheric Steam Dump Controllers to 1005 psig in order to maintain the RCS at no-load Tavg. Per the AOP-10 Background, the Main Steam Isolation Valves are shut at Step 3 to ensure an excessive cooldown does not occur, which eliminates use of the steam dumps.

Learning Objective:

STATE the alternate control station locations for operating equipment outside the control room to safely conduct a plant shutdown.
(055.03.LP1275.004)

23. 2017 NRC 023/EPE/074EK1.08/2.8/3-SPK/RO/NEW/CSP-ST.0/043.03.LP1997.002
Given the following:

- The crew is implementing EOP-1, Loss of Reactor or Secondary Coolant
- One reactor coolant pump is running
- Containment conditions are NOT adverse

Which of the following set of conditions would result in an ORANGE path based on the Core Cooling Critical Safety Function Status Tree?

A. RCS Pressure 2350 psig
Core Exit Thermocouple 620°F
Wide Range RVLIS 45 ft

B. RCS Pressure 2350 psig
Core Exit Thermocouple 620°F
Wide Range RVLIS 35 ft

C. RCS Pressure 1605 psig
Core Exit Thermocouple 580°F
Wide Range RVLIS 45 ft

D.✓ RCS Pressure 1605 psig
Core Exit Thermocouple 580°F
Wide Range RVLIS 35 ft

RO Tier 1 Group 2

Source: New

Question History:
None

K/A:
074EK1.08 Inadequate Core Cooling
Knowledge of the operational implications of the following concepts as they apply to the Inadequate Core Cooling: **Definition of subcooled liquid.**
(Imp 2.8/3.1)

Cognitive Level:
Comprehension 3-SPK: The student must recall the definition of subcooling and determine the amount of subcooling for each scenario, then recall the core cooling status tree, and based on subcooling, and wide range reactor vessel level determine which will result in an orange path.

Reference:

CSP-ST.0 Unit 1, Critical Safety Function Status Trees, Rev 9, Figure 2

Proposed reference to be provided to the applicants during examination:

Steam Tables

Justification:

The saturation temperature for 1647 psig is 610°F, and with core exit thermocouples (CETs) reading 580°F the RCS subcooling would be 30°F, The required subcooling if CETs are less than 1200°F is greater than 35°. With low subcooling, and one reactor coolant pump running, the wide range reactor vessel level is required to be greater than 40 ft.

A **INCORRECT:** Subcooling is greater than 35°F. Plausible because the student must determine the amount of subcooling in order to determine if a value of wide range RVLIS is required and the level given will result in a yellow path.

B **INCORRECT:** Subcooling is greater than 35°F. Plausible because the student must determine the amount of subcooling in order to determine if a value of wide range RVLIS is required and the level given will result in an orange path.

C **INCORRECT:** Subcooling is less than 35°F. Plausible because this RVLIS level will result in a yellow path.

D **CORRECT:** See above explanation.

Learning Objective:

State the RED path and ORANGE path entry conditions for CSP-C.1 and C.2 from memory.
(043.03.LP1997.002)

24. 2017 NRC 024/EPE/E01EK2.1/3.3/2-DR/RO/BANK/EOP-0.0/031.02.LP0405.011
Given the following:

- Unit 1 has experienced a transient outside of Containment that has resulted in a Reactor trip and Safety Injection
- All MSIVs are closed
- The crew took actions to address a cooldown in EOP-0, Reactor Trip or Safety Injection, then transitioned to EOP-1, Loss of Reactor or Secondary Coolant
- The crew is now currently performing the actions of EOP-0.0 Rediagnosis to determine if there are any faulted or ruptured steam generators
- The Balance of Plant Operator observes the following steam generator indications

	NR Level	**Pressure**	**MSL Radiation**	**AFW Flow**
SG 'A'	60% and ↑	900 psig and →	0.120 μCi/cc	0 gpm
SG 'B'	5% and ↓	700 psig and ↓	0.006 μCi/cc	0 gpm

Which one of the following correctly describes the status of the SGs?

A. 'A' faulted and 'B' faulted

B. 'A' ruptured and 'B' ruptured

C. 'A' faulted and 'B' ruptured

D.✓ 'A' ruptured and 'B' faulted

RO Tier 1 Group 2

Source: Bank

Question History:
2010 South Texas Project NRC RO 39

K/A:
E01EK2.1 Rediagnosis
Knowledge of the interrelations between the (Reactor Trip or Safety Injection/Rediagnosis) and the following: **Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.** (Imp 3.3/3.5)

Cognitive Level:
Comprehension 2-DR: The student must recall the indications used in EOP-0.0 to distinguish faulted and ruptured steam generators and apply that knowledge to diagnose a condition with one of each in progress.

Reference:
EOP-0.0 Unit 1, Rediagnosis, Rev 13, Steps 1 and 3

Proposed reference to be provided to the applicants during examination:
None

Justification:

With the crew having transitioned from EOP-0 to EOP-1, the actions to address the cooldown in EOP-0 step 8 were lowered AFW flow to zero on both SGs. Also, the crew failed to make a correct diagnosis while in EOP-0 and recognized their error after transitioning to EOP-1.

A **INCORRECT:** SG 'A' is plausibly faulted if you see that pressure is not rising while the level is rising. SG 'B' is faulted.

B **INCORRECT:** SG 'A' is ruptured. SG 'B' is plausibly ruptured if you believe that the 0.006 μCi/cc is indicating tube failure, and/or misunderstand the other indications.

C **INCORRECT:** SG 'A' is plausibly faulted if you see that pressure is not rising while the level is rising. SG 'B' is plausibly ruptured if you believe that the 0.006 μCi/cc is indicating tube failure and/or misunderstand the other indications.

D **CORRECT:** SG 'B' is faulted as indicated by decreasing pressure and decreasing level. SG 'A' is ruptured as indicated by the NR level continuing to rise with no AFW flow. The S/G pressure is relatively stable due to the isothermal layer developed.

Learning Objective:

Given access to the Site Specific simulator or specific plant conditions, IDENTIFY the proper Control Board indications for implementing procedural steps in the EOPs.
(031.02.LP0405.011)

25. 2017 NRC 025/EPE/E15EK3.4/2.9/2-DR/RO/NEW/1-SOP-RH-002/055.01.LP0272.001
Given the following:

- Unit 1 is in MODE 4
- Train A RHR is aligned for cooling
- Train B RHR is aligned for low head safety injection
- RCS is solid
- RCS pressure is 300 psig and STABLE
- RCS temperature is 295°F and LOWERING SLOWLY

Operating which of the following from the Control Room will cause flooding of Containment and challenge core cooling?

A✓ SI-851A, 1P-10A RHR Pump Suction from Cont Sump B

B. SI-852B, 1P-10B RHR Pump RV Injection valve

C. SI-857A, 1HX-11A RHR HX Outlet to 1P-15A SI Pump Suction

D. SI-871A, 1P-14A Cont Spray Pump RHR Suction

RO Tier 1 Group 2

Source: New

Question History:
None

K/A:
E15EK3.4 Containment Flooding
Knowledge of the reasons for the following responses as they apply to the (Containment Flooding): **RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.** (Imp 2.9/3.0)

Cognitive Level:
Comprehension 2-DR: The student must understand the initial conditions, determine which of valves will cause a operate based on initial conditions, and if operated, will cause the containment flooding and a loss of core cooling.

Reference:

1-SOP-RH-002, Residual Heat Removal System Operation, Rev 10, Section 3.6.1

Proposed reference to be provided to the applicants during examination:

None

Justification:

1-SOP-RH-002, 3.6.1 states “Opening of SI 851A, P-10A RHR Pump Suction From Cont Sump B, OR SI 851B, P-10B RHR Pump Suction From Cont Sump B will cause a rapid depressurization of the RCS and Flooding of the containment 8’ level. Depending on the temperature of the RCS, this could result in injury to personnel in the containment in the vicinity of Sump B, If RCPs are operating, RCP Seal Damage may result, and RCS/Cavity inventory lose will result.” This valve is caution tagged with a note which states "SM permission required to reposition."

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible, this valve can be operated given the initial conditions, and if the student has a misconception that there is no check valve in the line, will assume flow from the RCS causing a loss of core cooling ability. Leakage via this path is the typical ECA-1.2, LOCA Outside Containment seen during training.

C **INCORRECT:** Plausible, per 1-SOP-RH-002, 3.6.3, Opening SI-857A, HX-11A RHR HX Outlet to P-15A SI Pump Suction or SI-857B, HX-11B RHR HX Outlet to P-15B SI Pump Suction, could cause a rapid depressurization of the RCS and a loss of RCS/Cavity inventory, but given the initial conditions, the 210 psig interlock will keep the valve from opening remotely. This valve is caution tagged with a note which states "SM permission required to reposition."

D **INCORRECT:** Plausible, per 1-SOP-RH-002, 3.6.2, Opening SI-871A, P-14A Cont Spray Pump RHR Suction From HX-11A RHR HX or, SI-871B, P-14B Cont Spray Pump RHR Suction From HX-11B RHR HX could result in spraying down the containment, a rapid depressurization of the of the RCS and loss of RCS/Cavity inventory, but given the initial conditions, the open interlock (SI-870A is required to be closed), will keep the valve from operating remotely. This valve is caution tagged with a note which states "SM permission required to reposition."

Learning Objective:

DESCRIBE the procedures which govern the operations associated with taking the plant from Hot Standby Operation to Cold Shutdown Operation. Description should include significant prerequisites, precautions, and notes associated with each operating procedure requiring consideration by Licensed Operators. (055.01.LP0272.001)

26. 2017 NRC 026/EPE/E03EA1.3/3.7/1-P/RO/BANK/EOP-1.2/031.012.LP0435.001

Given the following:

- A Loss of Coolant Accident has occurred on Unit 1
- The crew has transitioned to EOP-1.2, Post LOCA Cooldown and Depressurization
- Prior to the step which depressurizes the RCS to refill the Pressurizer, there is a note stating the upper head region may void during depressurization

Which of the following conditions could cause the upper head region to void AND what would be the result if voiding occurs?

	CAUSE	RESULT
A.✓	Reactor Coolant Pumps NOT RUNNING	Rapidly rising Pressurizer Level
B.	RCS is not COUPLED to the Steam Generators	Loss of RCS Subcooling
C.	RCS is not COUPLED to the Steam Generators	Rapidly rising Pressurizer Level
D.	Reactor Coolant Pumps NOT RUNNING	Loss of RCS Subcooling

Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "'not coupled' means the Steam Generators are not removing heat from the RCS."

RO Tier 1 Group 2

Source: Bank

Question History:
2011 Callaway NRC RO 25

K/A:
E03EA1.3 LOCA Cooldown and Depressurization
Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization): **Desired operating results during abnormal and emergency situations.** (Imp 3.7/4.1)

Cognitive Level:
Knowledge 1-P: The student must recall the note in the procedure.

Reference:

EOP-1.2 Unit 1, Post LOCA Cooldown and Depressurization, Rev 33, Step 9 and Step 9 Note

BG-EOP-1.2, Background Document Post LOCA Cooldown and Depressurization, Rev 27, EOP Step 9-Note 1

Proposed reference to be provided to the applicants during examination:

None

Justification:

The note prior to step 9 of EOP-1.2, states ‘The upper head region may void during RCS depressurization if RCPs are NOT running. This will result in a rapidly rising PZR level’.

A **CORRECT:** See above explanation.

B **INCORRECT:** The cause and result are both incorrect. Plausible as the steam generators are used for cooldown, and if unavailable would contribute to a loss of subcoling, and reducing RCS pressure will in fact lower subcooling due to lowering RCS pressure without lowering RCS temperature.

C **INCORRECT:** The cause is incorrect and the result is correct. Whether steam generators are or are not coupled to the RCS will not cause voiding in the head, lack of flow from a RCP will cause that.

D **INCORRECT:** The cause is correct and result is incorrect. Plausible as the RCPs not running is the cause, and a loss of subcooling is a direct result of the depressurization. The result of the voiding will not be a loss of subcooling, but an increase in PZR level.

Learning Objective:

Given access to the Site Specific Simulator, ASSESS the following:

a. Safety Injection System response to a small break LOCA
b. Incore Thermocouple response to changing core conditions
c. Residual Heat Removal System response to a Safeguards Actuation signal
d. Safety Injection system response to a large break LOCA
e. Pressurizer level and pressure responses during LOCAs
f. Subcooling Margin following Safety Injection actuation

(031.02.LP0435.001)

27. 2017 NRC 027/003/E08EA2.2/3.5/1-B/RO/BANK/BG-EOP-0/031.02.LP0405.011
Given the following:

- Unit 1 was operating at Rated Thermal Power near the Beginning of Life (BOL)
- A Reactor trip occurred
- The crew has entered EOP-0, Reactor Trip or Safety Injection
- Step 5, 'Perform ATTACHMENT A, Automatic Action Verification, while continuing with this procedure' is in progress.

Which of the following statements describes the basis for the verification of feedwater isolation in Attachment A, Automatic Action Verification?

A. To minimize feed flow to reduce the likelihood of thermal shock to SG components.

B. To minimize a faulted steam generator becoming masked from the cooldown caused by excessive feedwater flow.

C. To prevent excessive FW flow to intact SGs which would cause MTC to add positive reactivity and return to criticality.

D✓ To prevent excessive RCS cooldown that could aggravate the initiating plant transient if it were a steam line break.

RO Tier 1 Group 2

Source: Bank

Question History:
2012 Surry NRC RO 27

K/A:
E08EA2.2 Pressurized Thermal Shock
Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock): **Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.**
(Imp 3.5/4.1)

Cognitive Level:
Knowledge 1-B: The student must recall the reason for the actions in EOP-0, in order to maintain operation of the plant within the limitations of the license.

Reference:

BG-EOP-0, Unit 1, Reactor Trip or Safety Injection, Rev 42, EOP Step A1

Proposed reference to be provided to the applicants during examination:

None

Justification:

From BG-EOP-0, Step A1, 'The main feedwater system is isolated on a FW Isolation signal to prevent uncontrolled filling of any steam generator and the associated excessive RCS cooldown which could aggravate the transient, especially if it were a steamline break.'

A **INCORRECT:** Plausible if the student has the misconception that excess Feedwater when combined with auxiliary Feedwater could cause a thermal shock due to the cooldown the flow is causing.

B **INCORRECT:** Plausible as the basis for the faulted generator identification step defines uncontrolled as not under the control of the operator, and incapable of being control by the operator using available equipment. This would fit the first portion of the definition, until isolated the feed would cause pressure/level changes that are not under control of the operator.

C **INCORRECT:** Plausible because actions are taken to control an excess cooldown. Several actions are performed to control the excessive cooldown, including limiting AFW flow to the steam generators.

D **CORRECT:** See above explanation.

Learning Objective:

LIST the major actions accomplished by each of the following Emergency Procedures:

a. EOP-0
b. EOP-0.0
c. EOP-0.1
d. EOP-1.1

(031.02.LP0405.003)

28. 2017 NRC 028/SYS/003K6.14/2.6/1-P/RO/MODIFIED/OP 4B/051.01.LP0125.005

Given the following:

- Unit 2 is in MODE 5, performing RCS fill and vent after an outage during which the Steam Generator tubes were drained
- LTOP is in service
- RCS pressure is 250 psig
- VCT Pressure is 18 psig
- RCP Oil Lift pump has been running for 3 minutes
- RCP Lift Pressure Amber light is LIT

Which of the following choices states why the Reactor Coolant Pump starting requirements of OP 4B Reactor Coolant Pump Operation are NOT met?

A. The VCT pressure requirement has not been met

B✓ The minimum RCS pressure requirement has not been met

C. The required run time for the Oil Lift pump has not been met

D. The required interlock for the lift oil system has not been met

RO Tier 2 Group 1

Source: Modified

Question History:
None

K/A:
003K6.14 Reactor Coolant Pump System (RCPS)
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: **Starting requirements.** (Imp 2.6/2.9)

Cognitive Level:
Knowledge 1-P: The student recall the reactor coolant pump starting requirements and procedural precautions.

Reference:
OP 4B, Reactor Coolant Pump Operation Rev 61, Section 3.17

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given the following:

- *Preparations are being made to start 1P-1B, B RCP, per OP 4B, Reactor Coolant Pump Operation*
- *1P-74B, 1P-1B Oil Lift Pump, has been running for 4 minutes*
- *1P-74B discharge pressure is 325 psig*
- *The amber light 1P-1B oil lift pump on 1C04 is* ***NOT LIT***

Which of the following choices states whether the 1P-1B RCP can be started per OP 4B, Reactor Coolant Pump Operation, and the reason why?

A. No, the B RCP cannot be started because the oil lift pump has not been running long enough.

B. No, the B RCP pump cannot be started because the oil pressure interlock has not been satisfied.

C. Yes, the B RCP pump can be started because the the minimum oil lift pump run time has been satisfied.

D. Yes, the pump can be started because the oil pressure interlock and the minimum oil lift pump run time have both been satisfied.

Proposed Answer: B

Justification:

Given the initial condition, the pressure requirement of 325-375 psig in the RCS is not met.

A **INCORRECT:** VCT pressure is required to be 15 psig. Plausible if the student incorrectly recalls the requirement because the given value is below the normal of 20 psig.

B **CORRECT:** See above explanation.

C **INCORRECT:** Minimum time for the oil pump to run it 2 minutes. Plausible if the student incorrectly recalls the requirement

D **INCORRECT:** The amber light indicates the interlock is met. Plausible if the student incorrectly recalls the requirement because the light is normally off when the RCP is running.

Learning Objective:

DESCRIBE the procedures which govern operation of the Reactor Coolant Pump System. Description should include significant prerequisites, precautions, and notes associated with each operating procedure requiring consideration by Licensed or Auxiliary Operators.
(051.01.LP0125.005)

29. 2017 NRC 029/SYS/003A3.03/3.2/2-DI/RO/BANK/684J741 SH 2/055.03.LP3718.001

Given the following:

- The unit is operating at full power with the following stable initial conditions:
  - Pressurizer level is 45.6%
  - Charging line flow is 29.5 gpm
  - VCT level is 53%
  - Lab seal ∆P is 39" and 41" for RCP A and B

- 30 minutes later the following conditions are noted with no operator action:
  - Pressurizer level is 45.6%
  - Charging line flow is 31 gpm
  - VCT level is 53%
  - Lab seal ∆P is 31" and 33" for RCP A and B

Which of the following is occurring?

A. Leak on the seal return line

B.✔ Seal injection filter plugging

C. Leak on the seal injection line

D. Charging pump relief leaking by

RO Tier 2 Group 1

Source: Bank

Question History:
2009 PBNP NRC RO 28

K/A:
003A3.03 Reactor Coolant Pump System (RCPS)
Ability to monitor automatic operation of the RCPS, including: **Seal D/P.**
(Imp 3.2/3.1)

Cognitive Level:
Comprehension 2-DI: The student must understand the initial conditions, and then determine how events will affect CVCS, and if it matches the conditions.

Reference:
684J741 Sh 2, Chemical and Volume Control P&ID, Rev 76
684J741 Sh 3, Chemical and Volume Control P&ID, Rev 17

Proposed reference to be provided to the applicants during examination:
None

Justification:
Given the conditions of a constant pressurizer level (with an increase in charging, but no corresponding lowering of VCT) combined with lowering Lab seal ΔP, the cause is the seal injection filter is plugging.

A **INCORRECT:** Plausible as charging has increased, but Lab seal ΔP would be constant and VCT level would lower.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible, as this would cause leak like symptoms and cause both charging flow and Lab seal ΔP to lower.

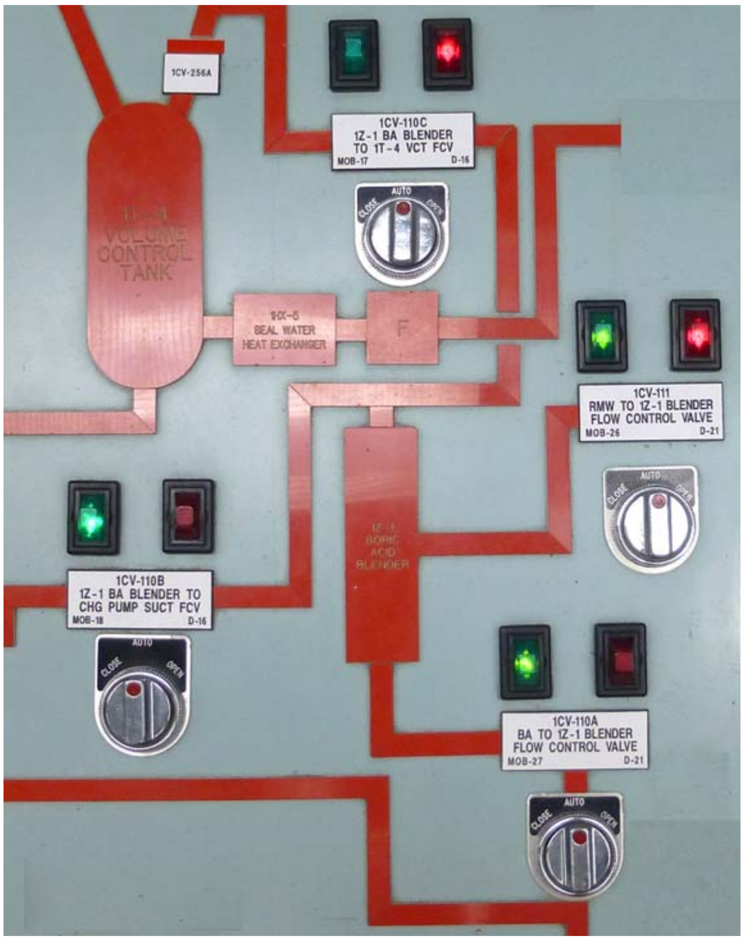
D **INCORRECT:** Plausible as this would cause similar symptoms, and cause both charging flow and Lab seal ΔP to lower and VCT to rise.

Learning Objective:
LIST the symptoms/indications of CVCS malfunctions.
(055.03.LP3718.001)

30. 2017 NRC 030/SYS/004G2.4.31/4.2/2-DR/RO/BANK/OP 5B/051.02.LP0079.009

Given the following:

- The crew is performing an Alternate Dilution per Attachment C of OP 5B, Blender Operation/Dilution/Boration
- The desired quantity of Dilution Flow and Reactor Makeup Water Flow have been set on the appropriate controllers
- The Reactor Makeup Mode Selector Switch is placed in ALT DIL
- The Reactor Makeup Control Switch is placed in START



Which valve failed to operate?

A. 1CV-110A

B✔ 1CV-110B

C. 1CV-110C

D. 1CV-111

RO Tier 2 Group 1

Source: Bank

Question History:
2016 Prairie Island NRC RO 29

K/A:
004G2.4.31Chemical and Volume Control System
Knowledge of annunciator alarms, indications, or response procedures.
(Imp 4.2/4.1)

Cognitive Level:
Comprehension 2-DR: The student must recall the alternate dilution procedure, what affect the switch operation will have, and what the system response will be.

Reference:
OP 5B, Blender Operation/Dilution/Boration, Rev 41, Attachment C, Alternate Dilution, Step 4.12

Proposed reference to be provided to the applicants during examination:
None

Justification:

When the system is in a standby lineup, the Reactor Makeup Mode Selector switch in AUTO and Reactor Makeup switch in AUTO, the valves are in the following positon, 1CV-110A is OPEN, 1CV-110B is SHUT, 1CV-110C is SHUT and 1CV-111 is SHUT. When the Reactor Makeup Mode Selector switch is taken to ALT DIL and then the Reactor Makeup switch is positioned to START the valves should move to the following positions 1CV-110A is SHUT, 1CV-110B is OPEN, 1CV-110C is OPEN and 1CV-111 is THROTTLED. 1CV-110B is in the incorrect position.

A **INCORRECT:** Plausible if the student incorrectly believes this valve should be open during a boration.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible if the student incorrectly believes this valve should be shut during a boration.

D **INCORRECT:** Plausible if the student incorrectly believes this valve should be shut during a boration.

Learning Objective:

IDENTIFY and DESCRIBE the Control Room controls, alarms, and indications associated with the Chemical Volume and Control System, including:

a. Location and function of component and/or system operating controls and control stations

b. Alarming locations and response to major system and component alarms

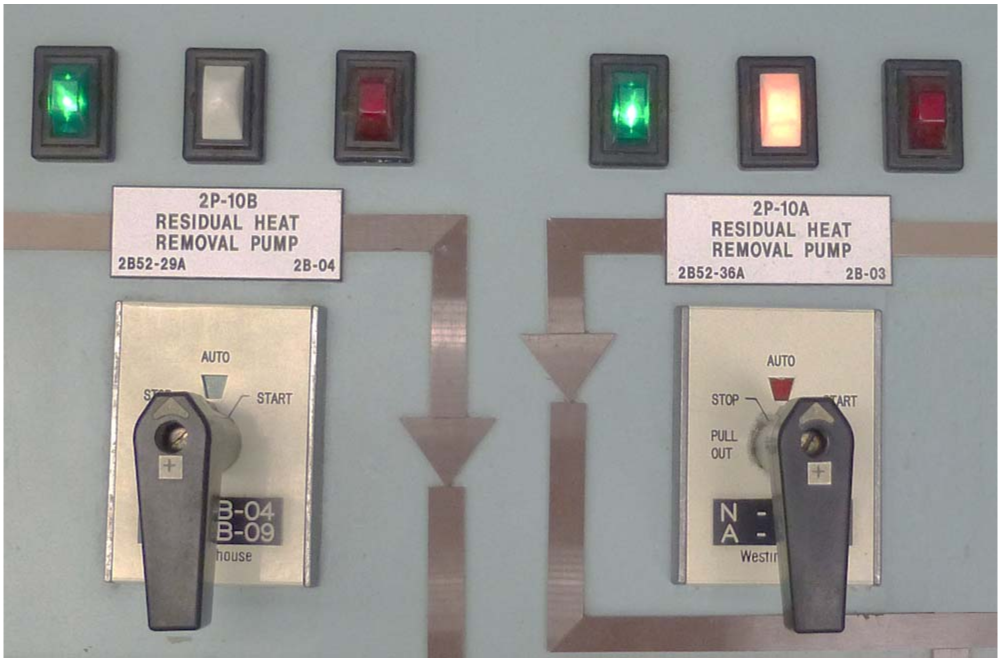
c. Plant, system, and component conditions or permissive required for Control Room operation

d. Setpoints associated with major system alarms and/or interlocks

(051.02.LP0079.009)

31. 2017 NRC 031/SYS/005A4.01/3.6/1-B/RO/MODIFIED/499B466 SH 337/051.03.LP0069.006
Given the following:

- Unit 2 is in MODE 4
- OP 7A, Placing Residual Heat Removal System in Operation is being performed
- When 2P-10A is started, the indication below is observed:



- Annunciator 2C03 2D 4-8, RHR LOOP FLOW LO is **LIT**

Which of the following describes the status of 2P-10A?

A. Not running due to Control Switch at C-45, Alternate Shutdown Control Panel being in PULLOUT

B. Not running due to blown control power fuses

C. Tripped on bus differential

D.✓ Tripped on overcurrent

RO Tier 2 Group 1

Source: Modified

Question History:
2008 Diablo Canyon NRC 32

K/A:
005A4.01 Residual Heat Removal System (RHRS)
Ability to manually operate and/or monitor in the control room: **Controls and indication for RHR pumps.** (Imp 3.6/3.4)

Cognitive Level:
Knowledge 1-I: The student must recall what causes specific light/switch position combinations on the RHR pump controls in the control room.

Reference:
499B466 Sh 337, Elementary Wiring Diagram Residual Heat Removal Sys Residual Heat Removal Pumps 2P-01A & 2P-10B, Rev 22
499B466 Sh 478, Elementary Wiring Diagram Alternate Supply 2P-01A/B Breaker B52-58B, Rev 3

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given the following plant conditions:

- *Unit is at 100% power*

During a surveillance test on RHR Pump B, the following control switch indications are observed after the pump is started:
Red Light - OFF / Green Light - ON / Amber Light - ON

- *Locally at the breaker:*
- *Red light - OFF / Green Light – ON / White Light – ON / Blue Light - OFF*
- *Annunciator 00-050A, RHR Pump Trouble, is in alarm*

Which ONE of the following describes the status of RHR Pump B?

A. Tripped on overcurrent

B. Tripped on differential

C. Not running due to loss of control power

D. Not running after a local stop

Proposed Answer: A

Justification:

With the red flag showing, which indicates the switch was taken to start, and the white light lit, an overcurrent condition has occurred.

A **INCORRECT:** Plausible as local operations of the switch will cause the pump to run/not run depending on the switch position, but will not be a cause for the white light to illuminate.

B **INCORRECT:** Plausible, as a loss of control power will not allow the pump to start, but will not cause the white light to illuminate.

C **INCORRECT:** Plausible as this is a type of trip, but a bus differential would not cause the white light to illuminate.

D **CORRECT:** See above explanation.

Learning Objective:

IDENTIFY and DESCRIBE the Control Room controls, alarms, and indications associated with the Residual Heat Removal System, including:

a. Location and function of component and/or system operating controls and control stations

b. Alarming locations and response to major system and component alarms

c. Plant, system, and component conditions or permissives required for Control Room operation

d. Setpoints associated with major system alarms and/or interlocks

e. Major Alarm and Actuation Setpoints

(051.03.LP0069.006)

32. 2017 NRC 032/SYS/006K3.02/4.3/1-B/RO/BANK/110E017/051.03.LP0066.001

The emergency core cooling system, operates in three modes in the event of a loss of coolant accident (LOCA). Those modes are passive accumulator injection, active safety injection and residual heat removal recirculation.

During the passive safety injection phase, the safety injection system provides ___(1)___.

If any of these modes of safety injection are not available during a design basis Loss of Coolant Accident, peak cladding temperature could exceed the 10 CFR 50.46 criteria of ___(2)___.
(Assume no operator action)

A✔ (1) cold leg injection
(2) 2200°F

B. (1) reactor vessel injection
(2) 2200°F

C. (1) cold leg injection
(2) 2700°F

D. (1) reactor vessel injection
(2) 2700°F

Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "the question is asking where passive injection is directed."

RO Tier 2 Group 1

Source: Bank

Question History:
2011 Turkey Point NRC RO 31

K/A:
006K3.02 Emergency Core Cooling System (ECCS)
Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: **Fuel.** (Imp 4.3/4.4)

Cognitive Level:
Knowledge 1-B: The student must recall the method of active injection and what the PCT limit of 10 CFR 50.46 is should one of those modes be unavailable.

Reference:

10 CFR 50.46, Acceptance criteria for emergency core cooling system for light-water nuclear power reactors, section (b)(1)
110E017 Sh 1, Safety Injection System P&ID, Rev 59
541F091 Sh 1, Reactor Coolant System P&ID, Rev 54

Proposed reference to be provided to the applicants during examination:

None

Justification:

The method of passive safety injection is cold leg injection. The Peak Cladding Temperature limit per 10 CFR 50.46 is 2200°F.

A **CORRECT:** See above explanation.

B **INCORRECT:** Peak cladding temperature is correct. Plausible as Reactor vessel injection is a method of active injection.

C **INCORRECT:** Peak cladding temperature is incorrect. Plausible if the student recalls the Peak Cladding Temperature for a non-LOCA limit for ZIRLO®..

D **INCORRECT:** Peak cladding temperature and reactor vessel injection are incorrect. Plausible as Reactor vessel injection is a method of active injection and if the student recalls the Peak Cladding Temperature for a non-LOCA limit for ZIRLO®.

Learning Objective:

DESCRIBE the function and/or purpose, design bases, and operating characteristics of the Safety Injection System.
(051.03.LP0066.001)

33. 2017 NRC 033/SYS/006G2.2.36/3.1/3-SPK/RO/NEW/OI 100/051.03.LP0066.009

Given the following:

- Both units are operating at Rated Thermal Power
- G02, Emergency Diesel Generator is out of service for maintenance overhaul
- G01, Emergency Diesel Generator is aligned to supply standby emergency power to 1A05 and 2A05, 4160 VAC Safeguards Busses
- The crew is currently raising level in 1T-34A, Safety Injection Accumulator, using 1P-15A, Safety Injection pump
- 2P-15A, Safety Injection pump is in PULL-OUT
- An event occurs on Unit 2 which results in an automatic Safety Injection

What action(s) will the CO filling the accumulator take?

A. No actions are required

B. Stop 1P-15A only

C. Leave 1P-15A running
Immediately place 2P-15A in AUTO

D✓ Stop 1P-15A
Start 2P-15A after the safeguards sequence is complete

RO Tier 2 Group 1

Source: New

Question History:
None

K/A:
006G2.2.36 Emergency Core Cooling System (ECCS)
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
(Imp 3.1/4.2)

Cognitive Level:
Comprehension 3-SPK: The student must understand the initial conditions and procedural/LCO requirements which cause the opposite unit SI pump to be declared inoperable when there are degraded power sources during this maintenance. Then the student must determine the actions required based on the Unit 2 SI.

Reference:
OI 100 Unit 1, Adjusting SI accumulator Level and Pressure, Rev 12, Section 3.8, and Step 5.2.7.d

Proposed reference to be provided to the applicants during examination:
None

Justification:

The student must recall, when one EDG is aligned to supply 1A05/2A05, then if filling an accumulator, the opposite unit will have the 'A' SI pump declared inoperable, and placed in pullout no longer meeting the LCO. If an SI happens on that opposite unit, the running SI pump is required to be stopped, and after the safeguards sequence is complete, the SI pump that was declared inoperable is loaded on to the safeguards bus.

A **INCORRECT:** Plausible if the student has the misconception that the unit 2 SI pump is still in AUTO, and there are still 2 trains of ECCS available for that unit, and no information about off-site power has been given, therefore the power available is more than adequate to have 3 SI pumps running.

B **INCORRECT:** Plausible if the student has the misconception that the unit 2 SI pump is still in AUTO, and there are still 2 trains of ECCS available for that unit, and recalls the fact that if off site power is lost, diesel loading will be unnecessarily challenged.

C **INCORRECT:** Plausible if the student recalls that the unit 2 pump will be in pullout. This action will provide 2 trains of ECCS to unit 2, and no information about off-site power has been given, therefore the power available is more than adequate to have 3 SI pumps running.

D **CORRECT:** See above explanation.

Learning Objective:

DESCRIBE the procedures which govern operation of the Safety Injection System. Description should include significant prerequisites, precautions, and notes associated with each operating procedure requiring consideration by Licensed and Non Licensed Operators.
(051.03.LP0066.009)

IDENTIFY and DISCUSS the Technical Specifications associated with Safety Injection System components, parameters, and operation to include:
a. Limiting Conditions for Operation (LCO)
b. LCO Applicability
(051.03.LP0066.007)

34. 2017 NRC 034/SYS/007G2.4.20/3.8/1-P/RO/BANK/BG-EOP-3/031.02.LP0441.012

Given the following:

- Unit 1 was operating at Rated Thermal Power
- A Steam Generator Tube Rupture occurred
- A Loss of Off-Site power occurs concurrently with the manual Safety Injection
- The crew is preparing to depressurize the RCS using a PZR PORV to minimize break flow and refill the PZR

- The caution prior to this step states:
  'The PRT may rupture if a PZR PORV is used to depressurize the RCS. This may result in abnormal containment conditions.'

Which of the following statements describes the basis of this caution?

A✓ The PRT rupture disk may fail before RCS pressure is reduced to ruptured SG pressure. This will result in increasing Containment radiation and humidity. The crew should continue recovery of the ruptured steam generator.

B. The PRT rupture disk may fail before RCS pressure is reduced to ruptured SG pressure. This will result in increasing Containment radiation and humidity. The crew will need to address the Loss of Coolant indications immediately if this occurs.

C. Cycling of the PZR PORV should be minimized to avoid failure of the PRT rupture disc. Do not use the PORV if PRT rupture disc failure is imminent. The crew will need to transition to a different recovery procedure if rupture disc failure is imminent.

D. Cycling of the PZR PORV should be minimized to avoid failure of the PRT rupture disc. Use of Auxiliary Spray is preferred over use of a PORV. If unable to depressurize using Auxiliary Spray the crew will need to transition to a different recovery procedure.

RO Tier 2 Group 1

Source: Bank

Question History:
2009 Seabrook RO 34

K/A:
007G2.4.20 Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of the operational implications of EOP warnings, cautions, and notes. (Imp 3.8/4.3)

Cognitive Level:
Knowledge 1-P: The student must recall the basis of this caution.

Reference:
BG-EOP-3, Background Steam Generator Tube Rupture, Rev 36,
EOP Step 18-Caution-1

Proposed reference to be provided to the applicants during examination:
None

Justification:

If a PORV is used to depressurize the RCS, steam will be discharged to the PRT and depending on the initial conditions of the primary, the rupture disc may fail prior to depressurizing the RCS. When that occurs conditions in the containment will change, i.e., containment humidity and radiation levels. These are also symptoms of a Loss of Coolant Accident. The caution is provided to alert the operation of the more likely source of these indications and to continue unless directed by subsequent steps.

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible as the rupture disc may fail prior to depressurization, but these are not addressed unless dictated by subsequent procedure steps or foldout page items.

C **INCORRECT:** Plausible as caution 2 of step 18 does warn of cycling of the PORV to minimize discharge to the PRT, but the basis of this is to minimize the potential of valve failure.

D **INCORRECT:** Plausible, as the next step address the use of Aux Spray, as an RNO for the PZR PORV step.

Learning Objective:

STATE the basis for the priority in selecting an RCS pressure reduction mode during a Steam Generator Tube Rupture.
(031.02.LP0441.012)

35. 2017 NRC 035/SYS/008K1.04/3.3/3-SPK/RO/MODIFIED/AOP-9B/051.06.LP0084.002

Given the following:

- Unit 1 and unit 2 are operating at Rated Thermal Power
- The normal lineup for HX-12B, Component Cooling Water heat exchanger was restored after replacement of CC-721B, HX-12B CCW Heat Exchanger Relief
- Chemistry is in the process of sampling the RCS
- Annunciator 2C03 2D 3-6 2T-12 CC SURGE TANK LEVEL HIGH OR LOW alarms

Which of the following indications would be consistent for the specified component cooling leak?
(2RE-217, CC Water Liquid Monitor)

	Specified Leak Location	**Indications**
A.	2HX-11B, RHR Heat Exchanger	-Unit 2 Surge Tank Level RISING -Auto charging pump speed RISING -2RE-217 reading RISING
B.	CC-721B, HX-12B CCW Heat Exchanger relief lifts	-Unit 2 Surge tank level LOWERING -Slow rise of all CCW temperatures -2RE-217 reading STABLE
C.✓	2HX-3A, Non-Regenerative Heat Exchanger	-Unit 2 Surge Tank Level RISING -2RE-217 reading RISING -VCT level LOWERING SLOWLY
D.	2HX-14C, Reactor Hot Leg Sample Heat Exchanger	-Unit 2 Surge Tank level LOWERING -All RCS sampled values lower than expected due to sample dilution -Sample temperature hotter than expected for current valve positions

RO Tier 2 Group 1

Source: Modified

Question History:
2011 Kewaunee Audit RO 36

K/A:
008K1.04 Component Cooling Water System (CCWS)
Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: **RCS, in order to determine source(s) of RCS leakage into the CCWS.** (Imp 3.3/3.3)

Cognitive Level:
Comprehension 3-SPK: The student must use knowledge of the physical connections of the CCW system with other plant systems and the normal parameters of those systems to determine the correct specified location with the correct indications.

Reference:
AOP-9B Unit 2, Component Cooling System Malfunction, Rev 25, Step B7
RMSASRB CI 1RE-217, Radiation Monitoring System Alarm Setpoint & Response Book Channel Information Sheet, CC Water Liquid Monitor Unit 1, Rev 10, High Alarm Section

Proposed reference to be provided to the applicants during examination:
None

Original Question
Given the following:

- *The unit has been operating at 100% Rated Thermal Power for 100 days.*
- *Chemistry is in the process of obtaining a sample from Reactor Coolant System.*

Which of the following indications would be consistent for the specified Component Cooling System Leak Location?

Specified Leak Location	*Indications*
A Excess Letdown Heat Exchanger	*- Rise in Tavg* *- Reduction in the expected Auto Makeups occurring*
B. RHR 'A' Heat Exchanger	*- R-17, Comp Cooling Liquid Monitor, elevated Counts.* *- Rise in charging flow*
C. Letdown Heat Exchanger	*- LD-10, Letdown Pressure, Valve, closes to maintain outlet pressure of the Letdown Heat Exchanger* *- Component Cooling Surge Tank level is rising.*
D. RC Hot Leg Sample Heat Exchanger	*- Temperature of Component Cooling water from the Hot Leg Sample Heat Exchanger is higher than expected* *- All RCS sample result values are lower than expected*

Proposed Answer: B

Justification:
Given the initial conditions, all systems with the exception of the CCW heat

exchangers are in a normal lineup. A leak from the non-regenerative heat exchanger will cause a rise in the surge tank, a rise in rad monitor counts, a lowering of VCT level as it makes up for a loss of RCS (or a small coolant leak) and letdown flow will lower and then stabilize due to part of the from downstream of the orifices going to the CCW system instead of letdown.

A **INCORRECT:** This would not cause surge tank level to rise, but actually lower, based on RHR not being in-service, as this would be leakage out of the system. Plausible these indications would be correct if the RHR system was in service.

B **INCORRECT:** This would not cause a lowering surge tank level, but rise in surge tank. The rest of the indication would be indicative of a leak out of the system. Plausible as this mimics the symptoms of a relief lifting on HX-12B heat exchanger.

C **CORRECT:** See above explanation.

D **INCORRECT:** If a leak was in this location under the current condition, it would be a leak into the CCW system. Surge tank would rise, and there would be no appreciable effect on sampled values. Plausible, as these symptoms would be true if the RCS was depressurized.

Learning Objective:

DRAW and DISCUSS a one-line diagram of the Component Cooling Water System similar to Component Cooling Water (AV1137). Discussion of this drawing should include system flowpaths, major components, and interfaces with other major systems:

a. Surge Tank
b. CCW Pumps
c. CCW Heat Exchangers
d. Components supplied
1. Residual Heat Removal Heat Exchangers
2. Residual Heat Removal Pump cooling
3. Reactor Coolant Pump cooling
4. Excess Letdown Heat Exchanger
5. Seal Water Heat Exchanger
6. Non-Regenerative Heat Exchanger
7. Containment Spray Pump cooling
8. Safety Injection Pump cooling
9. Radwaste CCW (Unit 2 only)
10. Sample Heat Exchangers
11. Waste Gas Compressors (Unit 1 only)

(051.06.LP0084.002)

36. 2017 NRC 036/SYS/008K4.01/3.1/2-DI/RO/BANK/PBN LP0084/051.06.LP0084.004

Given the following:

- Unit 1 is operating at Rated Thermal Power
- Component Cooling Pumps are aligned as shown:

- An electrical perturbation causes the following to occur concurrently:
  - A Sudden Pressure relay actuation occurs on 1X04, Low Voltage Station Auxiliary Transformer
  - 1B03, 480 VAC Safeguards Bus supply breaker tripping open

Which of the following shows the correct indications for the Component Cooling Pumps 2 minutes after the electrical perturbation?
(Assume no operator action)

Question continued on next page

36. 2017 NRC 036/SYS/008K4.01/3.1/2-DI/RO/BANK/PBN LP0084/051.06.LP0084.004

Question continued from previous page

A.✓ 

B. 

C. 

D. 

RO Tier 2 Group 1

Source: Bank

Question History:
2012 Ginna NRC RO 15

K/A:
008K4.01 Component Cooling Water System (CCWS)
Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: **Automatic start of standby pump.** (Imp 3.1/3.3)

Cognitive Level:
Comprehension 2-DI: The student must understand the initial conditions, and the event which occurred, then determine what the lockout means, and how the CCW pumps will react, and what the final line up will be concerning pump indications.

Reference:
PBN LP0084 Lesson Plan, Component Cooling Water System, Rev 18, Section II.C.2.e
883D195 Sh 9, Safeguards Sequence Logic Drawing, Rev 19
STPT 8.1, Auxiliary Coolant System Setpoints: General Instrumentation Setpoint Document, Rev 9, 1(2)PIC-639
AOP-18A Unit 1, Train 'A' Equipment Operation, Attachment A, Rev 17, Attachment A Step A1

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions 'A' pump is the running pump. The running pump will "ride the bus" (the breaker will remain closed) with an undervoltage condition. The bus lockout will not cause the breaker to open. When the 'A' pump loses power, the system pressure will lower, when it gets to 35 psig, the standby pump will start. So both pumps will have the red light lit.

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible if the student has a misconception of the power supply, and determines that the pump does not lose power.

C **INCORRECT:** Plausible if the student has the misconception that the pump breaker will trip open on the bus lockout, as most other breaker do, and due to the loss of power combined with a lockout, the standby pump does not pick up without operator action, similar to an undervoltage and safety injection signal.

D **INCORRECT:** Plausible if the student has the misconception that the pump breaker will trip open on the bus lockout, as most other breakers do, and that the standby pump picked up due to the loss of system pressure.

Learning Objective:

DESCRIBE the interlocks associated with the Component Cooling Water System and its major components:
a. Vent valve to atmosphere (CC-17)
b. CC-769
c. RCP Pump CCW valves
d. RADWASTE CCW supply and return valves
e. Standby CCW Pump start
(051.06.LP0084.004)

37. 2017 NRC 037/SYS/010K1.06/2.9/2-DI/RO/BANK/OP 4A/055.02.LP0162.009
Given the following:

- Unit 1 is heating up following a forced outage
- The crew is performing OP 4A, Filling and Venting Reactor Coolant System
- The RCS is Solid
- RCS Pressure is 90 psig

Which of the following describes the operator action(s) necessary to raise RCS pressure in order to start the first Reactor Coolant Pump per OP 4A?

A. Adjust the controller output of 1HC-626 1HX-11A&B RHR HX Bypass Flow Ctlr, to raise flow bypassing the heat exchangers, causing RCS temperature to increase making RCS pressure rise

B. Adjust the controller output of 1HC-142, Charging Line Flow Controller, to raise flow to the RCP seals causing RCS pressure to rise

C.✓ Adjust the controller output of 1HC-135, LP Letdown Line Pressure Controller to raise letdown pressure causing RCS pressure to rise

D. Adjust the controller output of 1HC-133, RHR to Letdown Flow Controller, to lower RHR flow to letdown causing RCS pressure to rise

RO Tier 2 Group 1

Source: Bank

Question History:
2014 Surry NRC RO 13

K/A:
010K1.06 Pressurizer Pressure Control System (PZR PCS)
Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: **CVCS.** (Imp 2.9/3.1)

Cognitive Level:
Comprehension 2-DI: The student must understand the initial conditions, recall how pressure control is being maintained when the plant is solid, and then recall the method utilized to adjust RCS pressure to for RCP operations.

Reference:
OP 4A, Filling and Venting Reactor Coolant System, Rev 87, Step 5.22

Proposed reference to be provided to the applicants during examination:
None

Justification:
Given the initial conditions of solid plant control, charging is in manual at a constant value to give a fixed addition to the RCS. Adjusting 1HC-135 will cause a change in letdown flow from the RCS, causing pressure to rise or fall based on the amount of flow to letdown. Lowering the letdown flow will cause a rise in RCS pressure.

A **INCORRECT:** Plausible if the student misapplies prior training, as this will cause temperature to rise causing pressure to rise in a solid system with no automatic pressure control. The failure of this valve and controller is a topic of training and how that failure will affect RCS pressure while on RHR.

B **INCORRECT:** Plausible as charging is being controlled in manual to maintain a constant mass into the RCS, this will cause the pressure to increase but, RCP lab seals will decrease, and this is not the method utilized in the OP.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible as with a reduction in flow to letdown from RHR and the plant being solid, this will cause the pressure to increase but is not the method utilized in the OP

Learning Objective:
Given access to the Site Specific Simulator and the appropriate procedure(s), DEMONSTRATE the ability to control plant pressure during solid plant operations, including during performance of the Normal Leak Test per OP 1A. (055.02.LP0162.009)

38. 2017 NRC 038/SYS/012K2.01/3.3/1-I/RO/BANK/617F354 SH 5A/053.02.LP0273.003

Given the following:

- Unit 1 is at 5% Reactor Power
- 52/RTA, Reactor Trip Breaker 'A' is CLOSED
- 52/BYB, Reactor Trip Bypass Breaker 'B' is CLOSED

- Train 'A' power is lost to the Reactor Protection System

Which of the following is the automatic response of the Reactor Protection System?

	52/RTA, 'A' Reactor Trip Breaker	**52/BYB, 'B' Reactor Trip Bypass Breaker**
A.	OPEN	CLOSED
B.	CLOSED	OPEN
C.	CLOSED	CLOSED
D.✓	OPEN	OPEN

RO Tier 2 Group 1

Source: Bank

Question History:
2010 Surry NRC RO 37

K/A:
012K2.01 Reactor Protection System
Knowledge of bus power supplies to the following: **RPS channels, components, and interconnections.** (Imp 3.3/3.7)

Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "125 VDC power is lost to the Reactor Protection System Train 'A'."

Cognitive Level:
Knowledge 1-I: The student must recall the bus power supplies and apply an abnormal situation to that base knowledge.

Reference:
617F354 Sh 5A, Reactor Protection System Reactor Trip Breaker Switchgear Train A Unit 1, Schematic Diagram, Rev 5

Proposed reference to be provided to the applicants during examination:
None

Justification:

The ‘A’ train 125 VDC train supplies power to both the ‘A’ Reactor Trip Breaker and ‘B’ Reactor Trip Bypass Breaker UV coils, so both will be open.

A **INCORRECT:** Plausible if the student believes the ‘B’ Reactor Trip Bypass Breaker is powered from ‘B’ train 125 VDC power.

B **INCORRECT:** Plausible if the student believes the ‘A’ Reactor Trip Breaker is powered from ‘B’ train 125 VDC power.

C **INCORRECT:** Plausible if the student the believes ‘A’ Reactor Trip Breaker and ‘B’ Reactor Trip Bypass Breaker are not powered from the 125 VC system

D **CORRECT:** See above explanation.

Learning Objective:

STATE the power supplies for Reactor Protection and major components. (053.02.LP0273.003)

39. 2017 NRC 039/SYS/012K5.01/3.1/1-B/RO/BANK/TS B 3.3.1/057.02.LP3341.001

Which of the following reactor trips is required to be OPERABLE in BOTH MODE 2 and MODE 1, to provide protection against Departure from Nucleate Boiling (DNB) per Tech Spec, 3.3.1, Reactor Protection System (RPS) Instrumentation?

A. OPΔT

B.✔ OTΔT

C. Pressurizer low pressure

D. Reactor Coolant Flow-Low Two Loop

RO Tier 2 Group 1

Source: Bank

Question History:
2013 Catawba NRC RO 38

K/A:
012K5.01 Reactor Protection System
Knowledge of the operational implications of the following concepts as the apply to the RPS: **DNB.** (Imp 3.1/3.8)

Cognitive Level:
Knowledge 1-B: The student must recall which feature of RPS must be operable to provide protection against DNB.

Reference:
TS B 3.3.1, Bases for Reactor Protection System (RPS) Instrumentation, Rev 8, Applicable Safety Analyses, LCO and Applicability Section 5

Proposed reference to be provided to the applicants during examination:
None

Justification:

OTΔT trip function is provided to ensure that the design limit DNBR is met. In MODE 1 or 2 the OTΔT trip must be OPERABLE to prevent DNB.

A **INCORRECT:** Plausible as this trip function ensures that the allowable heat generation rate of the fuel is not exceeded.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible as this trip function ensures that protection is provided against violating the DNBR limit due to low pressure, but it is only required to be operable min MODE 1 above P-7.

D **INCORRECT:** Plausible as this trip function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow, but it is only required to be operable in MODE 1 above P-7.

Learning Objective:

IDENTIFY and DISCUSS the Technical Specifications associated with the following Instrumentation System components, parameters, and operation including Limiting Conditions for Operation (LCO), LCO Applicability, Action Conditions and Required Actions as they pertain to the following requirements:

a. Reactor Protection System Instrumentation
b. Engineered Safety Feature Actuation System Instrumentation
c. Post Accident Monitoring (PAM) Instrumentation
d. Loss Of Power Diesel Generator Start and Load Sequence Instrumentation
e. Control Room Emergency Filtration System Actuation Instrumentation
f. Boron Dilution Alarm

(057.02.LP3341.001)

40. 2017 NRC 040/SYS/013KK2.01/3.6/1-F/RO/BANK/883D195 SH 7/053.06.LP0486.002
Given the following:

- Unit 1 was operating at Rated Thermal Power
- A Loss of Coolant Accident resulted in the following:
  - The reactor tripped automatically
  - Safety Injection actuated automatically
  - Off-site power to Unit 1 was lost at the time of the Safety Injection actuation
- The crew is taking actions in the Emergency Operating Procedures and has reached a procedural step requiring SI to be reset. Currently:
  - RCS pressure is 1250 psig and STABLE
  - 1A-05, 4160 VAC Safeguards Bus voltage is 0 VAC
  - 1A-06, 4160 VAC Safeguards Bus voltage is 4160 VAC
- CO3 has just depressed and released the Unit 1 SI Reset pushbuttons on C01R

Based on the conditions above, what is the status of the Unit 1 Safeguards Reset Bypass Activated red lights on C01R?

	Train A	**Train B**
A.	Lit	Lit
B.✓	Not lit	Lit
C.	Lit	Not lit
D.	Not lit	Not lit

RO Tier 2 Group 1

Source: Bank

Question History:
None

K/A:
013K2.01 Engineered Safety Features Actuation System (ESFAS)
Knowledge of bus power supplies to the following: **ESFAS/safeguards equipment control.** (Imp 3.6/3.8)

Cognitive Level:
Knowledge 1-F: The student must understand the initial conditions and recall how safeguards power affects the reset of a safety injection signal.

Reference:
883D195 Sh 7, Safeguards Actuation Signals Logic Diagrams, Rev 25

Proposed reference to be provided to the applicants during examination:
None

Justification:
With no ‘A’ safeguards power available the MG-6 relay will not reposition to reset train ‘A’ components. Depressing the SI reset PB will momentarily make up the lock in reset circuitry causing the SI Train ‘A’ Reset Bypass Activated Light on C01R to light, but it will immediately clear due to the 1A05 undervoltage relay not allowing the lock in.

A **INCORRECT:** Plausible as this is expected response with power available to 1A05 and 1A06.

B **CORRECT:** See above explanation**.**

C **INCORRECT:** Plausible as this is the expected response with power lost to the ‘B’ train of safeguards power.

D **INCORRECT:** Plausible as this is the expected response with power available to both 1A05 and 1A06 and no SI signal present.

Learning Objective:
STATE the power supplies for the Engineered Safety Features Actuation System and its major components.
(053.06.LP0486.002)

41. 2017 NRC 041/SYS/013A1.01/4.0/2-DI/RO/BANK/BG-ECA2.1/031.02.LP0465.008

Given the following:

- ECA-2.1, Uncontrolled Depressurization of Both Steam Generators is being performed
- The crew has reduced feed flow to 50 gpm to each steam generator (SG)
- Attempts to isolate the SGs continue

Which of the following describes
(1) the expected plant response to the feed flow reduction
AND
(2) what actions would be taken to mitigate the effect?

A✔ (1) RCS hot leg temperature will eventually begin to rise due to the reduction of SG inventory
(2) the crew will then raise AFW flow while continuing in ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

B. (1) RCS hot leg temperatures will eventually begin to rise due to the reduction of SG inventory
(2) The crew will then enter CSP-H.1, Response to Loss of Secondary Heat Sink to establish the required amount of feed water flow to control the heatup.

C. (1) The SGs will eventually become completely depressurized due to inadequate secondary heat sink
(2) the crew will then enter EOP-2, Faulted Steam Generator Isolation to isolate the SG and control feed water flow.

D. (1) The SGs will eventually become completely depressurized due to inadequate secondary heat sink
(2) the crew will then enter CSP-H.1, Response to Loss of Secondary Heat Sink to establish the required amount of feed water flow to control the heatup.

RO Tier 2 Group 1

Source: Bank

Question History:
2007 Ginna NRC RO 12

K/A:
013A1.01 Engineered Safety Features Actuation System (ESFAS)
Ability to predict and/or monitor changes in parameters (to Prevent exceeding design limits) associated with operating the ESFAS controls including: **RCS pressure and temperature.** (Imp 4.0/4.2)

Cognitive Level:
Comprehension 2-DI: The student understand the initial conditions and what the reduction in feed water flow will do to the RCS temperature values, and then determine what mitigation strategies will be employed by the procedure.

Reference:
BG-ECA-2.1, Background Uncontrolled Depressurization of Both Steam Generators, Rev 36, EOP Step 2
ECA-2.1 Unit 1, Uncontrolled Depressurization of Both Steam Generators, Rev 42, Step 2

Proposed reference to be provided to the applicants during examination:
None

Justification:

A reduction of feed flow is to minimize any additional cooldown resulting from addition of feedwater, prevent SG tube dry out and minimize the water inventory in the SG. As SG pressure and steam flow rate lower, RCS hot leg temperature will stabilize and start rising. The operator controls feed flow or dumps steam to stabilize the RCS hot leg temperatures.

A **CORRECT:** See above explanation.

B **INCORRECT:** Correct plant response, incorrect mitigation. Plausible if the student has a misconception on the method of mitigation and when/how feed flow would be increased.

C **INCORRECT:** Plant response and mitigation are incorrect. Plausible if the student treats this condition similar to a faulted steam generator, where it will eventually depressurize, and E-2 would be the logical procedure path with a faulted steam generator.

D **INCORRECT:** Plant response and mitigation are incorrect. Plausible if the student treats this condition similar to a faulted steam generator, where it will eventually depressurize, and the method of mitigation of when/how feed flow would be increased.

Learning Objective:

Given appropriate conditions/parameters and access to the site specific Simulator, IMPLEMENT the following procedures for the specified conditions:

a. ECA-1.1 to respond to a loss of Containment Sump Recirculation
b. ECA-1.2 to respond to an intersystem LOCA
c. ECA-1.3 to respond to containment sump blockage
d. ECA-2.1 to respond to both Steam Generators being faulted

(031.02.LP0465.008)

42. 2017 NRC 042/SYS/022K3.01/2.9/3-SPK/RO/NEW/EOP-0.2/031.02.LP0407.005

Given the following:

- Unit 1 was operating at Rated Thermal Power
- A loss of off-site power occurs on Unit 1
- The crew has commenced a natural circulation cooldown per EOP-0.2, Natural Circulation Cooldown
- RCS hot leg temperature has lowered from 570°F to 560°F
- Subsequently power has been restored to 1B-01, 480V Non-safeguards bus

(1) Should 1W-3A, CTL Rod Drive Shroud Fan be restarted?
AND
(2) Why or why not?

A.✓ (1) Yes
(2) After the shroud fan is started, the pressure/temperature limit curve is less restrictive than with no shroud fans running

B. (1) Yes
(2) The cooldown cannot be completed without causing damage to the Control Rod Drive Mechanisms unless a shroud fan is running

C. (1) No
(2) Once the cooldown is commenced, starting a shroud fan will cause uneven cooling to the Control Rod Drive Mechanisms and cause possible damage

D. (1) No
(2) Once the cooldown is commenced, starting a shroud fan does not change the pressure/temperature limit curves or soak temperature requirements

RO Tier 2 Group 1

Source: New

Question History:
None

K/A:
022K3.01 Containment Cooling System (CCS)
Knowledge of the effect that a loss or malfunction of the CCS will have on the following: **Containment equipment subject to damage by high or low temperature, humidity, and pressure.** (Imp 2.9/3.2)

Cognitive Level:
Comprehension 3-SPK: The student understand the initial conditions and determine what actions are required concerning the shroud fan, and when the action has been determined, then determine the impact on subsequent action of the procedure.

Reference:
EOP-0.2 Unit 1, Natural Circulation Cooldown, Rev 33, Step 2, Figure 1 and 2
BG-EOP-0.2, Natural Circulation Cooldown, Rev 27, EOP Step 2

Proposed reference to be provided to the applicants during examination:
None

Justification:

Step 2 of EOP-0.2 is a continuous action step (CAS), where you start both cavity cooling fans and one control rod shroud fan. Based on the CAS, the fan should be started when power is restored. Based on Steps 6, 10, the P/T limits curve used is based on one or no shroud fan running, so after one is started, then a less restrictive curve can be used.

A **CORRECT:** See above explanation.

B **INCORRECT:** Starting the fan is correct, the reason is incorrect. Plausible if the student has a misconception about the subcooling requirements, one fan must be running for the entire cooldown in order to use the less restrictive required subcooling value (Step 12).

C **INCORRECT:** The fan should be started per step 2, and the impact is also incorrect. Plausible if the student does not recall that the Establish Reactor Vessel Cooling step is a continuous action step and has the misconception that damage will be cause due to the CRDMs based on starting a fan after none have been running. Uneven cooling will happen, but should not cause damage.

D **INCORRECT:** The fan should be started per step 2, and the impact is also incorrect. Plausible if the student does not recall that the Establish Reactor Vessel Cooling step is a continuous action step and has a misconception that once the cooldown is started, the limits will not change.

Learning Objective:

Given a loss of offsite power, DESCRIBE the effect on plant response to controlling natural circulation.
(031.02.LP0407.005)

43. 2017 NRC 043/SYS/2026K3.02/4.2/2-DI/RO/NEW/BG-EOP-1.3/051.03.LP0064.009
Given the following:

- Unit 1 was operating at Rated Thermal Power
- A Large Break Loss of Coolant Accident (LOCA) occurred
- Flow blockage from foreign material has prevented normal NaOH flow
- The crew has lined up for containment spray recirculation
- Spray Add Tank Level has remained at 65% and is STABLE

The flow blockage may have prevented the containment sump from reaching a minimum pH of ___(1)___ which is required to ___(2)___?

A✔ (1) 7.0
(2) maximize the retention of iodine

B. (1) 7.0
(2) minimize the effects of boron precipitation

C. (1) 10.0
(2) maximize the retention of iodine

D. (1) 10.0
(2) minimize the effects of boron precipitation

RO Tier 2 Group 1

Source: New

Question History:
None

K/A:
26K3.02 Containment Spray System (CSS)
Knowledge of the effect that a loss or malfunction of the CSS will have on the following: **Recirculation spray system.** (Imp 4.2/4.3)

Cognitive Level:
Comprehension 2-DI: The student must understand the malfunction, and the implications of that malfunction on the containment sump, then recall what the minimum pH required is and why it is required.

Reference:

BG-EOP-1.3, Background Transfer to Containment Sump Recirculation – Low Head Injection, Rev 36, EOP Step 5

Proposed reference to be provided to the applicants during examination:

None

Justification:

The flow blockage has prevented adequate NaOH addition to the sump, as EOP-1.3 wants a minimum of 17% of the spray additive tank added, and the tank level has remained stable. Spray additive is added to maintain the sump pH between 7-9.5. A pH of greater than 7.0 assures the iodine removed by the spray is retained in the sump. At step 39,immediately after the SI alignment to address boron precipitation concerns, a check is made to determine if adequate NaOH has been added to the sump.

A. **CORRECT:** See above explanation

B **INCORRECT:** The minimum value is correct, the reason is wrong. Plausible if the student has a misconception on the purpose of NaOH addition to the sump.

C **INCORRECT:** The minimum value is wrong, but reason is correct. Plausible if the student has a misconception on what the minimum value pH for the sump is (pH above 9.5 causes excessive corrosion).

D **INCORRECT:** Both the minimum value and reason are wrong. Plausible if the student has a misconception on what the minimum value pH is for the sump and the purpose of NaOH addition to the sump.

Learning Objective:

DESCRIBE the procedures which govern operation of the Containment Spray System. Description should include significant prerequisites, precautions, and notes associated with each operating procedure requiring consideration by Licensed and Non Licensed Operators.
(051.03.LP0064.009)

44. 2017 NRC 044/SYS/039K4.04/2.9/2-DI/RO/BANK/391-92/052.02.LP0051.004
Given the following:

- The unit was at Rated Thermal Power when an inadvertent reactor trip occurred
- The crew is stabilizing RCS temperatures per EOP-0.1, Reactor Trip Response
- OS-1-MOV and OS-2-MOV, Crossover Steam Dump Manifold Inlet, control switches are in AUTO
- Crossover Steam Dump status is as shown below:

Are these indications correct for the plant conditions, why or why not?
(Assume no operator actions)

A. Yes, OS-1-MOV and OS-2-MOV will remain open until the turbine rotor is <600 RPM.

B. Yes, OS-1-MOV and OS-2-MOV will remain open until the close pushbuttons are depressed during post trip response.

C.✓ No, OS-1-MOV and OS-2-MOV should have automatically shut due to the lowering crossover steam dump header pressure.

D. No, OS-1-MOV and OS-2-MOV should have automatically shut due to the direct turbine trip signal generated during the reactor trip.

RO Tier 2 Group 1

Source: Bank

Question History:
2009 PBNP NRC RO 42

K/A:
039K4.04 Main and Reheat Steam System (MRSS)
Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: **Utilization of steam pressure program control when steam dumping through atmospheric relief/dump valves, including T-ave. limits.** (Imp 2.9/3.1)

Cognitive Level:
Comprehension 2-DI: The student must understand effect of the turbine trip on the crossover steam dump system, and then determine of the displayed status is correct for the given conditions and time passage, and determine if the system is functioning correctly in auto control and why

Reference:
391-92, Crossover Steam Dump Logic, Rev 6
STPT 14.13, Secondary Systems: Crossover Steam Dump Setpoint Document, Rev 9, 1DPS5933(DV) and 1(2)PS-5930(1SV)

Proposed reference to be provided to the applicants during examination:
None

Justification:

MOV-1 and MOV-2 will go intermediate and then shut in approximately 2 minutes after the turbine trip, not due to the turbine trip but due to the lowering crossover steam dump header pressure.

A **INCORRECT:** The indications are not correct and the reason is also incorrect. Plausible because 600 is a value with an automatic feature associated with it (lift oil pump start after trip).

B **INCORRECT:** The indications are not correct and the reason is also incorrect. Plausible as this action is normally taken during testing and manual control during startup.

C **CORRECT:** See above explanation.

D **INCORRECT:** The indications are not correct and reason is also incorrect. Plausible as the valves should have gone shut, but not due to the turbine trip signal, but due to the lowering pressure.

Learning Objective:

STATE actuation setpoints and EXPLAIN effects of automatic actuations and Interlocks associated with the Crossover Steam Dump System. (052.02.LP0051.004)

45. 2017 NRC 045/SYS/059K4.16/3.1/1-I/RO/MODIFIED/STPT 14.8/052.05.LP0128.004

Given the following:

- Unit 1 is at Rated Thermal Power

Which of the following conditions or situations will result in an automatic trip of <u>ONE OR BOTH</u> Steam Generator Feed Pumps(SGFP)?
(Assume no operator action)

A. Manual Reactor Trip and LOW TAVG

B. 1P-27A, Heater Drain Tank Pump breaker trips

C.✓ SGFP lube oil pressure at 7 psig for 3 seconds

D. SGFP suction pressure at 160 psig for 90 seconds

RO Tier 2 Group 1

Source: Modified

Question History:
2010 Prairie Island NRC RO 40
2013 Comanche Peak NRC RO 16

K/A:
059K4.16 Main Feedwater (MFW) System
Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: **Automatic trips for MFW pumps.** (Imp 3.1/3.2)

Cognitive Level:
Knowledge 1-I: The student must recall the trip setpoints for the steam generator feedwater pumps.

Reference:
STPT 14.8, Secondary Systems: Lube Oil Setpoint Document, Rev 8,
1(2)PS-3643A9B)-1/2

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Unit 1 is at 100% power.

What will cause only ONE Main Feed Pump to trip?

A. 12 SG level at 70% N.R.

B. RCS pressure at 1770 psig.

C 11 Condensate Pump breaker trips.

D. Feedwater pump suction pressure at 220 psig.

Proposed Answer: C

Justification:

SGFP lube oil pressure less than 9 psig for greater than 1.5 seconds will cause a pump trip of the respective pump.

A **INCORRECT:** Plausible if the student has the misconception that the two when combined will cause a SGFP trip, because a reactor trip will initiate Feedwater Isolation, with the exception of tripping the SGFPs and Low TAVG is an input on the Safety Injection logics.

B **INCORRECT:** Plausible as this will cause SGFP suction to lower, but not to a value low enough to cause a SGFP trip. The LP Feedwater Heater Bypass Pressure controller will start to open mitigating the effects of the loss of the pump SGFP suction pressure.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible as this pressure value is a start permissive for suction pressure.

Learning Objective:

DESCRIBE the interlocks, automatic actuations, and permissives associated with major components of the Feedwater System.
(052.05.LP0128.004)

46. 2017 NRC 046/SYS/061K5.01/3.6/1-B/RO/MODIFIED/BG-EOP-1/052.05.LP0169.001

EOP-1, Loss of Reactor or Secondary Coolant, has the following as step 3:

****

3	**Check Intact S/G Level:**	
	a. S/G level - GREATER THAN [51%] 32%	a. Maintain total feed flow greater than or equal to 230 gpm until level is greater than [51%] 32% in at least one S/G.
	b. Control feed flow to maintain S/G level between [51%] 32% and 63%	b. **IF** level in any S/G continues to rise in an uncontrolled manner, **THEN** go to EOP-3, UNIT 1, STEAM GENERATOR TUBE RUPTURE.

****

The reason for the minimum AFW flow of 230 gpm is to . . .

A. maintain SG level in the narrow range during a large break LOCA.

B. recover from the initial shrink in SG water level during the reactor trip.

C. ensure a sufficient generator inventory to establish natural circulation on a loss of off site power.

D.✓ ensure a secondary heat sink for small and intermediate size LOCAs and secondary breaks.

RO Tier 2 Group 1

Source: Modified

Question History:
2012 Diablo Canyon NRC RO 15

K/A:
061K5.01 Auxiliary / Emergency Feedwater (AFW) System
Knowledge of the operational implications of the following concepts as the apply to the AFW: **Relationship between AFW flow and RCS heat transfer.**
(Imp 3.6/3.9)

Cognitive Level:
Knowledge 1-B: The student must recall the basis for the minimum required AFW flow while in EOP-1.

Reference:
BG-EOP-1, Loss of Reactor or Secondary Coolant, Rev 37, EOP Step 3

Proposed reference to be provided to the applicants during examination:
None

Original Question:
E-0, Reactor Trip or Safety Injection, requires the operator to ensure AFW flow is at least 435 gpm.

The basis for this minimum flow is to:

A. make up for the initial shrink in SG water level.

B. remove RCS decay heat.

C. maintain SG water level in the narrow range.

D. ensure a sufficient heat sink to initiate natural circulation.

Proposed Answer: B

Justification:

The purpose of the step is to ensure adequate feed flow or steam generator inventory to ensure a secondary heat sink for small and intermediate size LOCAs and secondary break accidents exists. To provide a positive static head of water to prevent primary to secondary leakage of a large LOCA.

A **INCORRECT:** Plausible as it is beneficial to maintain SG narrow range levels on span to ensure a positive static head, level will not be maintained in the narrow range during the beginning of the event as the SG will most likely shrink out of the narrow range, and a minimum feed of 230 gpm does not maintain that level, it maintains the heat sink requirement..

B **INCORRECT:** Plausible as this will happen on most reactor trips.

C **INCORRECT:** Plausible as this is what the EOP generically does by performing this step and one of the assumptions for most accidents is a loss of off site power.

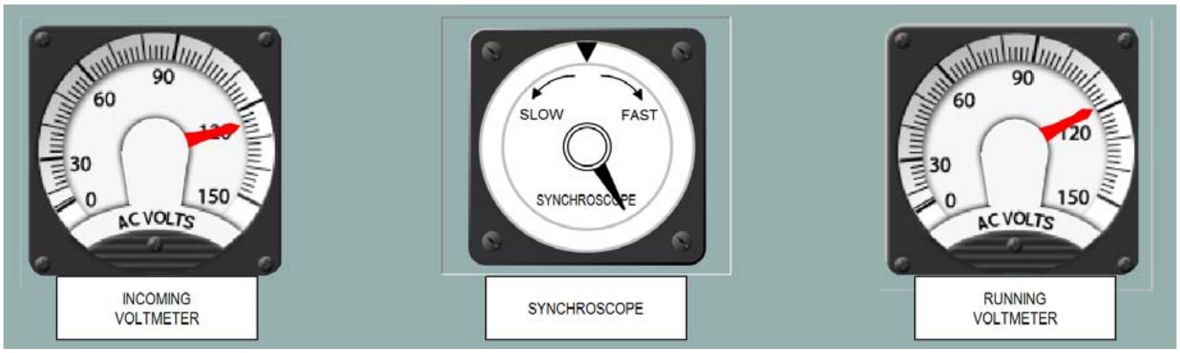
D **CORRECT:** See above explanation.

Learning Objective:

DESCRIBE the function and/or purpose, design bases, and operating characteristics of the Auxiliary Feedwater System and major components. (052.05.LP0169.001)

47. 2017 NRC 047/SYS/062A1.03/2.5/2-DI/RO/MODIFIED/TS 81/182.01.LP2519.001
Given the following:

- The Crew is performing TS-81, Emergency Diesel Generator G01 Monthly Test
- While paralleling G01 to 1A-05, the following conditions are noted:
  - Sync Selector Switch for 1A52-60, G01 to 1A-05 breaker, is ON
  - Synchroscope is rotating 10 RPM in the SLOW direction
  - Running and Incoming Voltmeter read as shown below:



What must the operator do to match voltage and make the Synchroscope turn 2 to 5 RPM in the FAST direction?

Go to ___(1)___ on the G01 Diesel Generator Voltage Regulator to equal or slightly exceed bus voltage.

Go to ___(2)___ on the G01 Diesel Generator Governor control switch to make Synchroscope rotate properly.

	(1)	(2)
A.	Raise	Raise
B.✔	Lower	Raise
C.	Raise	Lower
D.	Lower	Lower

RO Tier 2 Group 1

Source: Modified

Question History:
2005 PBNP NRC RO 48

K/A:
062A1.03 A.C. Electrical Distribution
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: **Effect on instrumentation and controls of switching power supplies** (Imp 2.5/2.8)

Cognitive Level:
Comprehension 2-DR: The operator must recall understand the initial conditions, what adjusting the governor control and voltage regulator will do the EDG, then determine which way and what to adjust to get the desired results which will meet the limitation and requirements to allow the source to be shifted.

Reference:
TS 81, Emergency Diesel Generator G01 Monthly Test, Rev 86, Step 5.29

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given the following:

- The Crew is performing TS-81, Emergency Diesel Generator G01 Monthly Test
- While paralleling G01 to 1A-05, the following conditions are noted:
  - Sync Selector Switch for 1A52-60, G01 to 1A-05 breaker, is ON
  - Running Voltmeter reads 123 volts
  - Incoming Voltmeter reads 118 volts
  - Synchroscope is rotating 10 RPM in the SLOW direction

What must the operator do to match voltage and make the Synchroscope turn 2 to 5 RPM in the FAST direction?

Go to ___(1)___ on the G01 Diesel Generator Voltage Regulator to match voltages. Go to ___(2)___ on the G01 Diesel Generator Governor control switch to make Synchroscope turn properly.

	(1)	(2)
A.	*Raise*	*Raise*
B.	*Lower*	*Raise*
C.	*Raise*	*Lower*
D.	*Lower*	*Lower*

Proposed answer: A

Justification:

Running Voltage is the voltage of the bus, in this case. Incoming Voltage is the EDG voltage. EDG voltage needs to be lowered to match bus, thus Lower voltage on G01 make Incoming match Running. If the synch scope is going FAST in the SLOW direction, the EDG is not turning fast enough, the governor adjust will need to be raised to make the scope turn slow in the fast direction.

A **INCORRECT:** Plausible if the student has the misconception of which is running and which is incoming or the correct action to be taken.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible if the student has the misconception of which is running and which is incoming or the correct action to be taken and on what the actions are necessary to have the sync scope rotate in the correct direction.

D **INCORRECT:** Plausible this is the correct action to be taken to adjust EDG voltage regulator and if the student has a misconception on the actions necessary to have the sync scope rotate in the correct direction.

Learning Objective:

Given access to the PBNP Site Specific Simulator, the trainee should be able to OPERATE the major components of the specified system in accordance with acceptable practices and procedures.
(182.01.LP2519.001)

48. 2017 NRC 048/SYS/063K1.02/2.7/1-I/RO/BANK/PBN LP0124/054.02.LP0124.018

Which of the following describes the impact on breaker operation capability with a loss of DC control power for 1B52-17B, 1X-14 Low Side Breaker on 1B04 safeguards bus?

1B52-17B ___(1)___ be opened remotely from the control room.

1B52-17B ___(2)___ trip open should a high fault current condition exist.

	(1)	(2)
A.	can	will
B.✔	can NOT	will
C.	can	will NOT
D.	can NOT	will NOT

RO Tier 2 Group 1

Source: Bank

Question History:
2009 PBNP NRC RO 50

K/A:
063K1.02 D.C. Electrical Distribution
Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems: **AC electrical system.** (Imp 2.7/3.2)

Cognitive Level:
Knowledge 1-I: The student must recall that the protective devices and how they function for this breaker.

Reference:
PBN LP0124, 480V Distribution, Rev 23, Section II.F.1.b.4) and II.F.1.d.3)

Proposed reference to be provided to the applicants during examination:
None

Justification:

With a loss of DC control power the 480v breaker cannot be opened remotely, however 480v safeguards breakers are equipped with an amptector device for overcurrent protection independent of DC control power

A **INCORRECT:** Plausible if the student has a misconception on how DC control power functions with this breaker.

B **CORRECT:** See above explanation.

C **INCORRECT:** Complete opposite answer, plausible if the student has a misconception on how DC control power functions with this breaker.

D **INCORRECT:** Plausible if the student does not recall the amptector device installed.

Learning Objective:

RECOGNIZE the response of the 480V Electrical Distribution System to a loss of DC Control Power.
(054.02.LP0124.018)

49. 2017 NRC 049/SYS/064K6.07/2.7/2-DI/RO/MODIFIED/TS 3.8.3/057.02.LP3344.002

Given the following:

- Unit 1 is in MODE 1
- Unit 2 is in MODE 4 cooling down in preparation for a forced outage
- The Unit 1 Turbine Hall AO was sent to investigate annunciator C02 D 2-6, G-01 EMERGENCY DIESEL
- The AO reports G-01, Emergency Diesel Generator (EDG) starting air left bank pressure is 120 psig and STABLE with K-4A, G-01 EDG Starting Air Compressor running constantly
- The Unit 2 Turbine hall AO has just notified the control room the discharge isolation valves for both G-04, EDG starting air compressors are in the shut position
- G-04 starting air bank pressure is 225 psig in both air banks

Which of the following correctly identifies the status of TS 3.8.3, Diesel Fuel Oil and Starting Air for G-01 and G-04?

	G-01	**G-04**
A.	met	met
B.	met	**not** met
C.✓	**not** met	met
D.	**not** met	**not** met

RO Tier 2 Group 1

Source: Modified

Question History:
None

K/A:
064K6.07 Emergency Diesel Generators (ED/G)
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: **Air receivers.** (Imp 2.7/2.9)

Cognitive Level:
Comprehension 2-RI: The student must determine which EDGs required to be operable based on the initial conditions, then apply the requirements of the LCO to the given situations and determine if the LCO is met or not.

Reference:

TS 3.8.3, Diesel Fuel Oil and Starting Air, Rev 3

TS B 3.8.3, Diesel Fuel Oil and Starting Air, Rev 6, Background, and LCO Sections

Proposed reference to be provided to the applicants during examination:

None

Original Question:

Given the following:

- *Both units are in MODE 1*
- *The Unit 2 Turbine Hall AO was sent to investigate an Emergency Diesel Generator alarm and has reported that G-04, Emergency Diesel Generator (EDG) starting air left bank pressure is 85 psig and slowly lowering*
- *The Unit 1 Turbine hall AO has just notified the control room that he found the discharge isolation valves for both G-01, EDG starting air compressors in the shut position*
- *Pressure is normal in both G-01 starting air banks*

Which of the following correctly identifies EDG OPERABILITY?

	G-01	***G-04***
A.	*OPERABLE*	*OPERABLE*
B.	*OPERABLE*	***not*** *OPERABLE*
C.	***not*** *OPERABLE*	*OPERABLE*
D.	***not*** *OPERABLE*	***not*** *OPERABLE*

Proposed Answer: B

Justification:

The student must determine which EDGs are required to be operable, and with unit 1 in mode 1 and unit 2 in mode 4, all EDGs are required to be operable. The minimum pressure for the starting air banks is 165 psig, with one of the starting air banks at less than this for G-01, 3.8.3 is NOT MET. For G-04, the starting air banks are at normal pressure, so they are greater than 165 psig AND not isolated from the EDG so TS 3.8.3 is MET (even though with the compressors isolated, they will not be automatically maintained at pressure).

A **INCORRECT:** G-01 is incorrect. Plausible as the EDGs have 2 air banks and only one is needed to start the EDG, and G-01 has one back at normal operating pressure.

B **INCORRECT:** Both G-01 and G-04 are incorrect. Plausible as the EDGs have 2 air banks and only one is needed to start the EDG, and G-01 has one back at normal operating pressure, and G-04 is in not in a normal lineup (the compressor is isolated from the air receivers) and the compressors cannot maintain the required pressure. Normally when a system is not in a normal lineup it is usually not meeting the LCO.

C **CORRECT:** See above explanation.

D **INCORRECT:** G-04 is incorrect. Plausible as G-04 is in not in a normal lineup (the compressor is isolated from the air receivers) and the compressors cannot maintain the required pressure. Normally when a system is not in a normal lineup it is usually not meeting the LCO.

Learning Objective:

Given specific plant conditions, ASSESS and APPLY Technical Specification requirements as appropriate.
(057.02.LP3344.002)

50. 2017 NRC 050/SYS/064A2.13/2.6/3-SPK/RO/NEW/OI 168/057.02.LP3344.002
Given the following:

- Both units are at Rated Thermal Power
- D28, DC Distribution Panel is de-energized due to a fault which caused the supply fuses to blow

Which of the following action(s) is/(are) required based on the loss of D28?

A. Station a Dedicated Operator at C-101, G-03 EDG Engine Gauge Control Panel with the duty of manually flashing the field for G-03

B✓ Remove G-03 EDG from service and align G-04, EDG to 1A-06, 4160 VAC Safeguards Bus

C. Align alternate supply DC Distribution Panel, D-40 within 1 hour or commence a Unit 1 shutdown

D. Start repairs on D28, G-03 is not affected by the loss

RO Tier 2 Group 1

Source: New

Question History:
None

K/A:
064A2.13 Emergency Diesel Generators (ED/G)
Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Consequences of opening auxiliary feeder bus (ED/G sub supply).
(Imp 2.6/2.8)

Cognitive Level:
Comprehension 3-SPK: The student must understand the initial conditions, determine what impact the event will have on the EDGs, and then based on that effect, determine the appropriate actions.

Reference:
OI 168, Emergency Diesel Generator Operability, Rev 23, Step 5.11.2

Proposed reference to be provided to the applicants during examination:
None

Justification:

Per procedure, both diesels (G03 and G04) cannot be lined up to the same DC panel, therefore one will be declared inoperable (G03 based on the initiating event) and the other will be aligned to both 4160 VAC Safeguards Buses.

A **INCORRECT:** Plausible as this is a possible use of a dedicated operation, but is not procedurally supported.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible if the student attempts to apply tech spec LCO 3.8.4, DC Sources – Operating to the situation.

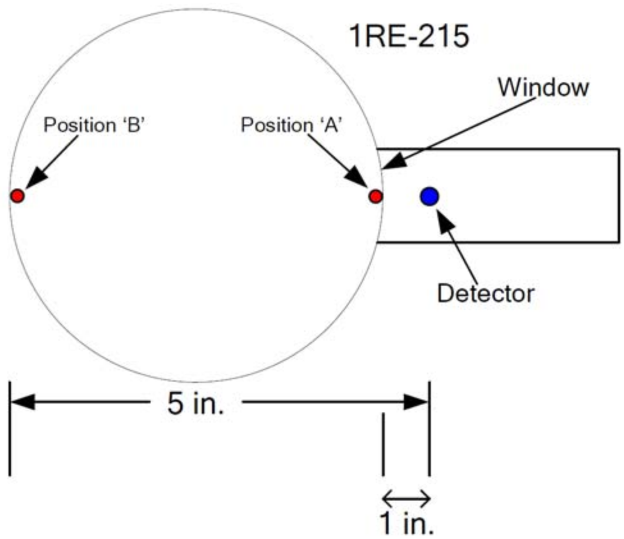
D **INCORRECT:** Plausible if the student has the DC power supplies backwards (thinks D28 supplies G-04).

Learning Objective:

Given specific plant conditions, ASSESS and APPLY Technical Specification requirements as appropriate.
(057.02.LP3344.002)

51. 2017 NRC 051/SYS/073K5.02/2.5/3-SPK/RO/BANK/PBN LP3946/HPI-01-LP3946.010

1RE-215, Condenser Air Ejector Noble Gas Monitor, consists of a detector mounted on one side of the air ejector pipe which is 4 inches in diameter (see below). The detector is located 1 inch from the inner wall of the pipe.



Compare the differences in radiation readings for a hot particle that passes along the side of the pipe closest to the detector (Position 'A' above) to the same hot particle that passes along the side of the pipe farthest away from the detector (Position 'B' above).

The radiation reading at Position 'A' will be ______ greater than the reading at Position 'B'.

A. 4 times

B. 5 times

C. 16 times

D✓ 25 times

RO Tier 2 Group 1

Source: Bank

Question History:
2009 Palisades NRC RO 50

K/A:
073K5.02 Process Radiation Monitoring (PRM) System
Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: **Radiation intensity changes with source distance.** (Imp 2.5/3.1)

Cognitive Level:

Comprehension 3-SPK: The student needs to understand and apply the formula for a point source to the example given and determine the change in intensity

Reference:
PBN LP3946, Nuclear Interactions, Rev 1

Proposed reference to be provided to the applicants during examination:
None

Justification:

$I_1D_1^2=I_2D_2^2$

Therefore dose is proportional to the square of the distance, so a count rate of 25dpm at 1 inch would be equivalent to a count rate of 1dmp at 5 inches.

A **INCORRECT:** Plausible as this is a linear proportionality which does not account for the 1 inch distance of the detector.

B **INCORRECT:** Plausible as this is a linear proportionality.

C **INCORRECT:** Plausible as this is a squared proportionality which does not account for the 1 inch distance of the detector.

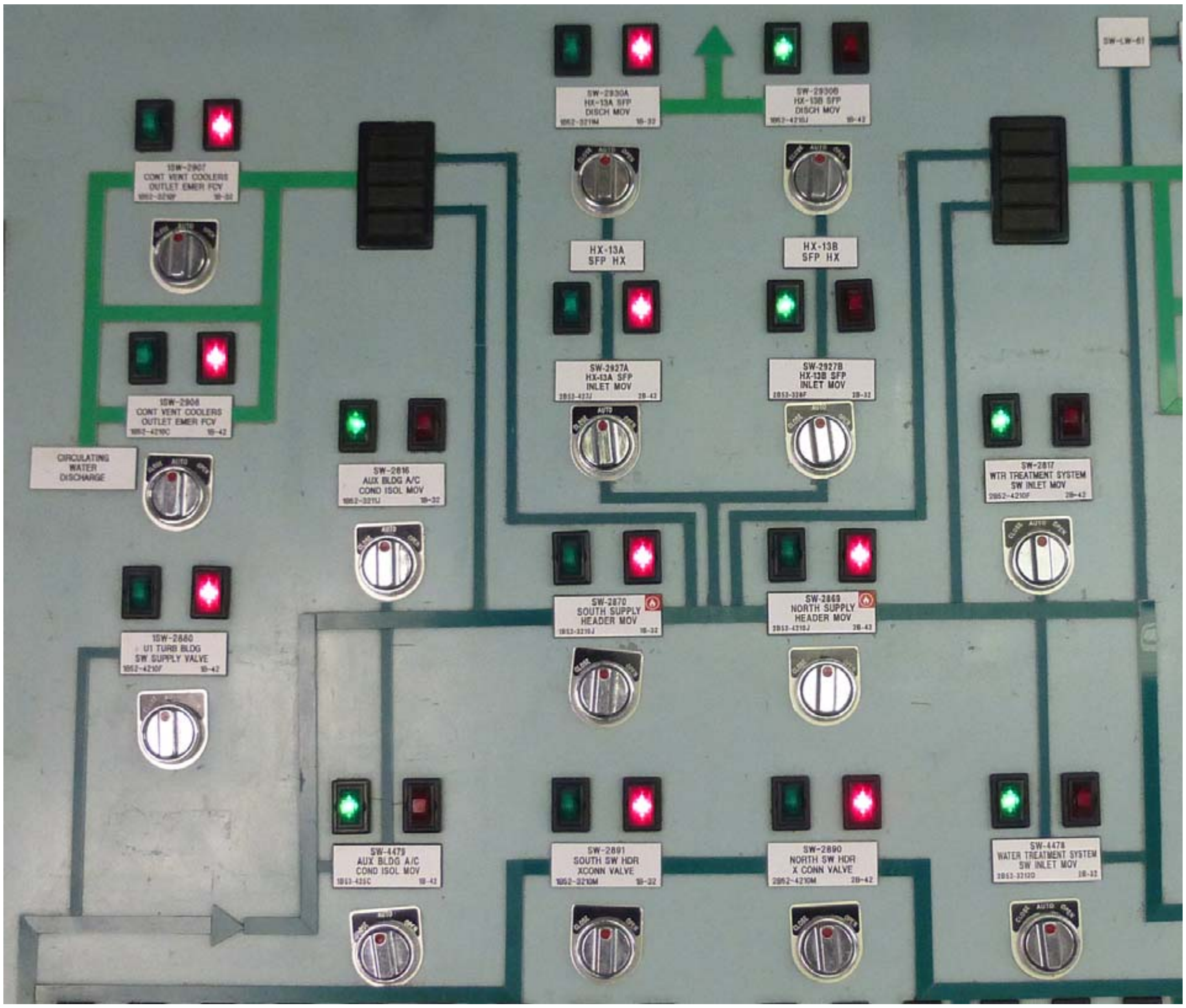
D **CORRECT:** See above explanation.

Learning Objective:
Given a point source or a line source, calculate the dose rate(s) at given distances.
(HPI-01-LP3946.010)

52. 2017 NRC 052/SYS/076A3.02/3.7/1-I/RO/NEW/883D195 SH 8/051.06.LP0086.008

Given the following:

- Unit 1 is at Rated Thermal Power
- Unit 2 has completed a refueling outage and is heating up
- A steam line rupture downstream of the Main Steam Isolation Valves occurs on Unit 1 causing a Safety Injection (SI)
- EOP-0, Attachment A, Automatic Action Verification step A15 is being performed, ‘Verify Service Water System Alignment’
- The status of the Service Water system is pictured below:



Which of the following identifies the status of the Service Water line up?
(Assume both trains of Safety Injection have activated)

Question continued on next page

52. 2017 NRC 052/SYS/076A3.02/3.7/1-I/RO/NEW/883D195 SH 8/051.06.LP0086.008

Question continued from previous page

A. All valves have functioned as designed, the lineup is as expected for the SI signal received

B. 1SW-2880, U1 Turb Bldg SW Supply Valve is not in the correct position, it is designed to close on any SI signal

C.✓ SW-2927A/SW-2930A HX-13A SFP Inlet/Disch MOVs are not in the correct position, they are designed to close on any SI signal

D. SW-2817/SW-4478, Water Treatment System SW Inlet MOVs are not in the correct positon, they are designed to close ONLY on a Unit 2 Safety Injection

RO Tier 2 Group 1

Source: New

Question History:
None

K/A:
076A3.02 Service Water System (SWS)
Ability to monitor automatic operation of the SWS, including: **Emergency heat loads.** (Imp 3.7/3.7)

Cognitive Level:

Knowledge 1-I: The student must recall the automatic actions of the Service Water system which occur during the SI sequence, which ensure the heat loads serviced be SW have the required amount of flow.

Reference:
883D195 Sh 8, Safeguards Sequence Logic Diagram, Rev 19

Proposed reference to be provided to the applicants during examination:
None

Justification:

On any SI the following valves are required to operate
Shut to isolate flow paths
SW-2927A/2927B/2930A/2930B, SFP heat exchanger isolations
SW-2816/4479, Aux building A/C condenser isolations
SW-4478/2817, water treatment
Open to provide flow
1SW-2907/2908, containment ventilation cooler outlet emergency FCVs

A **INCORRECT:** Plausible if the student has the misconception that only one SFP cooler receives an isolation signal.

B **INCORRECT:** Plausible if the student has the misconception that the turbine building isolation receives a closed signal on an SI and this was a previous design of this valve.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible if the student has the misconception that a since the valves are powered from Unit 2, they will only be affected by a Unit 2 safety injection.

Learning Objective:

ASSESS the response of the service water system to a Safeguards actuation. (051.06.LP0086.008)

53. 2017 NRC 053/SYS/078K2.01/2.7/2-DI/RO/BANK/MDB/052.06.LP0338.003

Given the following:

- Both units are operating at Rated Thermal Power
- K2A, Instrument Air (IA) compressor is in CONSTANT
- K2B, Instrument Air (IA) compressor is in AUTO
- K3A, Service Air (SA) compressor is in STBY
- K3B, Service Air (SA) compressor is RUNNING

- A loss of offsite power to **BOTH** units occurs
- All Emergency Diesel Generators start and load onto their respective buses

Which of the following describes the expected impact on the units due to air systems?
(Assume no operator action)

A✓ BOTH unit MSIV Air Pressure Low alarms will sound. None of the air compressors will be running.

B. NEITHER unit MSIV Air Pressure Low alarms will sound. All four air compressors will automatically start due to low system header pressure.

C. Unit 1 MSIV Air Pressure Low alarm will sound. K2B, IA compressor will automatically start and K3B, SA compressor will automatically re-start maintaining air pressure to Unit 2 MSIVs.

D. Unit 2 MSIV Air Pressure Low alarm will sound. K2A, IA compressor will automatically re-start and K3A, SA compressor will automatically start maintaining air pressure to Unit 1 MSIVs.

RO Tier 2 Group 1

Source: Bank

Question History:
None

K/A:
078K2.01 Instrument Air System (IAS)
Knowledge of bus power supplies to the following: **Instrument air compressor.**
(Imp 2.7/2.9)

Cognitive Level:

Comprehension 2-DI: The student must understand the event and how that event will affect both the air system and how the air system will affect the MSIVs. Knowledge of power supplies to air compressors and electric plant system response to LOOP as well as MSIV and the impact from the loss of air system is necessary.

Reference:
MDB 3.2.3 PANEL 1B04, 480V AC Unit 1, Rev 15
MDB 3.2.4 PANEL 2B04, 480 V AC Unit 2, Rev 14
MDB 3.2.5 PANEL 1B32, 480 V AC Motor Control Centers Unit 1, Rev 23
MDB 3.2.6 PANEL 2B42, 480 V AC Motor Control Centers Unit 2, Rev 21
499B466 Sh 366, Serv Air Comp K-3A Elementary Wiring Diagram, Rev 25
499B466 Sh 366A, Serv Air Comp K-3B Elementary Wiring Diagram, Rev 12
499B466 Sh 538A, Inst Air Comp K-2A Elementary Wiring Diagram, Rev 7
499B466 Sh 538B, Inst Air Comp K-2B Elementary Wiring Diagram, Rev 8

Proposed reference to be provided to the applicants during examination:
None

Justification:

Instrument air is a shared common system between the units. Air compressors strip off their respective buses on an undervoltage and do not automatically sequence back on when power is restored. Operator action is needed to reset the undervoltage stripping condition. Without operator action no IA or SA compressors will be running, and air systems are commons, so both MSIVs will be affected.

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible if the student has the misconception that air compressors will sequence on once power is restored and that the SA and IA are common shared systems.

C **INCORRECT:** Plausible if the student has the misconception that the auto control scheme starts the standby IA and or in-service SA compressor(s) when power is restored and the compressors are unit specific.

D **INCORRECT:** Plausible if the student has a similar misconception to that above.

Learning Objective:

STATE the power supply for the following Instrument and Service Air components:

a. Instrument and Service Air Compressors

b. Instrument Air Dryers and Dryer Bypass Solenoid valves

(052.06.LP0338.003)

54. 2017 NRC 054/SYS/103A2.04/2.6/2-DI/RO/MODIFIED/ARB 1C04 1A 1-3/055.03.LP2442.008
Given the following:

- Unit 1 is in MODE 6
- Core reload was started last shift and 10 fuel assemblies were loaded
- Source Range Nuclear Instrument baselines have been established and are:
    - N31 is 40 CPS
    - N32 is 45 CPS
    - N40 is 43 CPS
- After the fifteenth (15) fuel assembly is released in the core, annunciator 1C04 1A 1-3 SOURCE RANGE HIGH FLUX LEVEL AT SHUTDOWN **alarms**
- Source Range Nuclear instruments currently read:
    - N31 is reading 239 CPS
    - N32 is reading 120 CPS
    - N40 is reading 201 CPS

Which of the following identifies
(1) If the alarm is valid
AND
(2) Action(s) you are required to perform?

A. (1) Yes
(2) Remove the fuel assembly from the core for placement in the Spent Fuel Pool.

B✓ (1) Yes
(2) Make a Gaitronics announcement to evacuate all personnel from containment.

C. (1) Yes
(2) Notify the refueling crew of the alarm, and monitor conditions during the core reload closely.

D. (1) No
(2) Make a Gaitronics announcement that alarm was not valid and all personnel may remain at their work locations.

RO Tier 2 Group 1

Source: Modified

Question History:
None

K/A:
103A2.04 Containment System
Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Containment evacuation (including recognition of the alarm).** (Imp 3.5/3.6)

Cognitive Level:
Comprehension 2-DI: The student must validate the alarm based on the size of the rise in the SR readings (1/2 decade), and recall the actions based on the whether or not the alarm is valid.

Reference:
ARB 1C04 1A 1-3, SOURCE RANGE HIGH FLUX LEVEL AT SHUTDOWN, Rev 6, Step 6.2
STPT 9.1, Alarms and Operational Adjustments Setpoint Document, Rev 8, Section 1.1

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given the following:

Unit 1 is in MODE 6

- *Core re-load is in progress with a new assembly being loaded in the core*
- *Annunciator 1C04 1A 1-3 SOURCE RANGE HIGH FLUX LEVEL AT SHUTDOWN is LIT*
- *N31 is reading 195 CPS and rising with a stable SUR of 0.5 DPM*
- *N32 is reading 200 CPS and rising with a stable SUR of 0.6 DPM*
- *N40 is reading 180 CPS and rising with a stable SUR of 0.2 DPM*

As the Unit 1 Control Operator, what actions are you required to perform?

A. Stop fuel motion and notify the refueling crew to leave containment.

B. Remove the fuel assembly from the core for placement in the Spent Fuel Pool and commence boration.

C. Make a Gaitronics announcement to evacuate all personnel from containment and commence boration.

D. Make a Gaitronics announcement that the High Flux at Shutdown alarm was not valid and have all personnel re-commence work.

Proposed answer: C

Justification:

The alarm is valid based on an increase of 4x the baseline or 0.5 decade increase in reading (background) on SR 31 SR 40 has a 0.6 decade rise, but does not input into the alarm. The action required is to evacuate containment.

SR Meter	Initial	Final	Log initial	log final	LOG Difference	Final/ Initial
31	40	239	1.60	2.38	0.78	5.98
32	45	120	1.65	2.08	0.43	2.67
40	43	201	1.63	2.30	0.67	4.67

A **INCORRECT:** Plausible the alarm is valid, and this action could mitigate the event, but it not the required action.

B **CORRECT:** See above explanation

C **INCORRECT:** Plausible the alarm is valid, and if the student has a misconception of when an evacuation is warranted.

D **INCORRECT:** Plausible if the student has a misconception about the alarm and this action is correct for a non-valid alarm.

Learning Objective:

DESCRIBE the Symptoms and Mitigating Actions for the following Abnormal Operating Procedures:
(055.03.LP2442.008)

a. AOP 8B
b. AOP 8C
c. AOP 8F
d. AOP 8G
e. AOP 8H

55. 2017 NRC 055/SYS/103A4.06/2.7/1-I/RO/BANK/10-SOP-CONT-001/PBN HPI 02 LP016.009

Regarding the Containment Upper Personnel Hatch which of the following lineups show that the mechanical interlock is functioning properly?

A.✔ Outer door OPEN
Outer door equalizing valve OPEN
Inner door CLOSED
Inner door equalizing valve CLOSED

B. Outer door OPEN
Outer door equalizing valve OPEN
Inner door CLOSED
Inner door equalizing valve OPEN

C. Outer door CLOSED
Outer door equalizing valve OPEN
Inner door OPEN
Inner door equalizing valve CLOSED

D. Outer door CLOSED
Outer door equalizing valve CLOSED
Inner door OPEN
Inner door equalizing valve CLOSED

RO Tier 2 Group 1

Source: Bank

Question History:
2011 Farley NRC RO 59

K/A:
103A4.06 Containment System
Ability to manually operate and/or monitor in the control room: **Operation of the containment personnel airlock door.** (Imp 2.7/2.9)

Cognitive Level:
Knowledge 1-I: The student must recall the interlocks for the containment personnel hatches and how they function.

Reference:
1-SOP-CONT-001, Containment Airlock Operation and Entry, Rev 20, Attachment A Step 8

Proposed reference to be provided to the applicants during examination:
None

Justification:
The Containment Personnel Access Hatch doors designed to-ensure that Containment integrity is maintained during normal entry/exit of Containment. To provide integrity and ensure proper operation of the door a set of interlocks is installed for operation for the Door Equalizing valve, Door Latch, and Door opening action.
During Airlock door opening, the interlock sequence is
1) Equalizing valve opens to equalize pressure across door
2) door latch operates to unlatch door
3) door swings open.
The sequence is reversed for closure.
A key feature is that the equalizing valve always opens prior to the door opening, remains open the entire time the door is open, and closes after the door closes. In addition, only one door can be operated at a time. This ensures that there is no air to air breach of Containment during door operation.

A **CORRECT:** See above explanation.

B **INCORRECT:** Causes a containment integrity breach. Plausible as the student may think this demonstrates the proper operation of the equalizing valve to allow door operation.

C **INCORRECT:** Causes a containment integrity breach. Plausible as the student may think this demonstrates the proper operation of the equalizing valve to allow door operation.

D **INCORRECT:** This does not cause a containment integrity breach. Plausible as the student may think since there is no breach, this demonstrates proper operation.

Learning Objective:
Describe the operation of the containment airlocks.
(PBN HPI 02 LP016.009)

56. 2017 NRC 056/SYS/001A3.07/4.1/3-SPR/RO/NEW/ROD 1.3/055.01.LP0258.014

Given the following:

- Unit 1 was initially operating at Rated Thermal Power
- The crew lowered power by 25% per AOP-17A, Rapid Power Reduction, in response to a secondary system malfunction
- Immediately following the power reduction the following parameters were observed:
  - TAVG is 568.8°F
  - Bank D Group Demand is 160 steps
  - Reactor Power is 75%
  - Gallons of boric acid added is 304 gallons

Using the reference provided (ROD 1.3)

(1) Which of the following parameters is <u>NOT</u> as expected for the given power change

AND

(2) What is a reason for the difference?

(Assume no effect from xenon)

A. (1) TAVG is low
(2) Boration was excessive

B. (1) TAVG is low
(2) Rod insertion was excessive

C. (1) Rod Bank Demand is low
(2) Boration was excessive

D.✓ (1) Rod Bank Demand is low
(2) One rod is stuck at 220 steps

RO Tier 2 Group 2

Source: New

Question History:
None

K/A:
001A3.07 Control Rod Drive System
Ability to monitor automatic operation of the CRDS, including:
Boration/dilution. (Imp 4.1/3.7)

Cognitive Level:
Comprehension 3-SPR: The examinee must recall the normal Tavg program and calculate its expected value for the given power level. Understand the reactivity effects of boron, rods, and temperature, and the effect that each has on the other. Use the information given in the stem to determine the expected values of boration and rod height from ROD 1.3 and then analyze the given scenario using information obtained from a reference and their understanding of core reactivity balance to determine the discrepant parameter.

Reference:
ROD 1.3 U1C37, Reactivity Plan for Power Changes, Rev 14

Proposed reference to be provided to the applicants during examination:
ROD 1.3 U1C37 for Burnup Range (4000-5000) MWD/MTU

Justification:

TAVG at 75%(in stem) = (576-547)0.75 + 547 = 568.75

From ROD 1.3 U1C37 (4000-5000) MWD/MTU: The expected values for a 25% power reduction from the AOP-17A table are:
304 gallon of boration
176 steps expected final Bank D position

The given value of 160 steps is 16 steps lower than the Bank Demand expected from ROD 1.3. A rod stuck at 220 steps would cause a lower than assumed bank rod worth, requiring more rod insertion to produce the same reactivity effect.

A **INCORRECT:** TAVG is on target for the given power change. Plausible if the student has a misconception about the method of determining TAVG, because excessive boration would cause TAVG to be low.

B **INCORRECT:** TAVG is on target for the given power change. Plausible if the student has a misconception about the method of determining TAVG, because excessive rod insertion would cause TAVG to be low and rod motion is excessive for the scenario described in the stem.

C **INCORRECT:** Rod bank demand is correct, boration is not. Plausible if the student has a misconception of the method of determining the amount of boron necessary, with rods being lower than expected an inadequate boration would be correct.

D **CORRECT:** See above explanation.

Learning Objective:

Predict the response of the rod control system for changing plant conditions. (055.01.LP0258.014)

57. 2017 NRC 057/SYS/011K6.03/2.9/1-I/RO/NEW/883D195 SH 18/051.01.LP0457.006

Given the following:

- Unit 2 is in MODE 4, heating up at the end of a refueling outage
- All four groups of pressurizer back-up heaters are energized
- Group C and Group D heaters are being operated from their local control station, 2N-04 and 2N-11, in support of post-maintenance testing
- Group C and Group D heaters are in LOCAL and ON
- 1LT-427, Pressurizer Level (White) fails **LOW**

What is the state of each group of heaters following the instrument failure?

	GP A	GP B	GP C	GP D
A.	ON	ON	ON	ON
B.	ON	ON	OFF	OFF
C.✓	OFF	OFF	ON	ON
D.	OFF	OFF	OFF	OFF

RO Tier 2 Group 2

Source: New

Question History:
None

K/A:
11K6.03 Pressurizer Level Control System (PZR LCS)
Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: **Relationship between PZR level and PZR heater control circuit.** (Imp 2.9/3.3)

Cognitive Level:
Knowledge 1-I: The examinee must recall that the low pressurizer level heater cutout interlock will operate from a single pressurizer level channel and that the cutout is blocked while in local control.

Reference:

883D195 Sh 18, Pressurizer Pressure and Level Control Logic Diagrams, Rev 13
883D195 Sh 19, Pressurizer Heater Control Logic Diagrams, Rev 5

Proposed reference to be provided to the applicants during examination:

None

Justification:

From the logic diagrams, the low pressurizer level heater cutout interlock operates from a single pressurizer level channel. This cutout is blocked while is the heater is in local control. Initially all back-up heaters are energized, the instrument failure will actuate low pressurizer level heater cutout protection. Groups C and D which are in local control, will therefore remain ON, while Groups A and B which are not in local control will be OFF.

A **INCORRECT:** Plausible if the student has a misconception on which instrument has the low level cutout associated with it.

B **INCORRECT:** Plausible if the student has effect of the protection feature backwards, where locally controlled will de-energize and remote will stay energized.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible if the student has the misconception that the cutout will affect all heaters regardless of the local operation.

Learning Objective:

Identify and discuss the response to the following failures on the Pressurizer Pressure and Level control system. b. Pressurizer level failing low (051.01.LP0457.006)

58. 2017 NRC 058/SYS/015K4.07/3.7/2-RI/RO/BANK/0-SOP-IC-002 WHITE/053.01.LP2416.011

Given the following:

- Unit 1 is at 8% reactor power during a startup
- Intermediate Range drawer N-36 Level Trip switch is in the BYPASS position

What is the plant response to removal of N-36 CONTROL power fuses and the reason for the plant response?

A.✓ A trip will occur, the Level Trip Bypass function will be removed

B. A trip will occur, the Level Trip Bypass function is active only above P-10

C. A trip will NOT occur, the Level Trip switch is in the BYPASS position and power is less than P-10

D. A trip will NOT occur, the Level Trip switch is in the BYPASS position and the bypass function is NOT affected by removal of Control Power fuses

RO Tier 2 Group 2

Source: Bank

Question History:
2005 PBNP NRC RO 58

K/A:
015K4.07 Nuclear Instrumentation System (NIS)
Knowledge of NIS design feature(s) and/or interlock(s) provide for the following: **Permissives.** (Imp 3.7/3.8)

Cognitive Level:
Comprehension 2-RI: The operator must understand the initial conditions, determine what effect the pulling of fuses will have on the detector, and if that effect will be different when the Level Trip switch is in bypass. The operator must then determine what effect that will have on the plant.

Reference:
0-SOP-IC-001 WHITE, Routine Maintenance Procedure Removal Of Safeguards Or Protection Sensor From Service - White Channels, Rev 16, Attachment A, Section 11

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions and the removal of the CONTROL POWER fuses, a reactor trip will occur due to the trip NOT being blocked because power is less than the P-10 permissive.

A **CORRECT:** See the above explanation.

B **INCORRECT:** Trip will occur, but not for the reason listed. Plausible if the student incorrectly recalls when the function is bypassed

C **INCORRECT:** Plausible due to this is the case if power is greater than the P-10 permissive.

D **INCORRECT:** Bypass function is not affected by removal of instrument power fuses, BUT would be affected by the removal of the control power fuses. Plausible if the student incorrectly recalls which fuses affect the bypass function.

Learning Objective:

Describe the operation and setpoints for Interlocks, Permissives and Automatic actions associated with the Nuclear Instrumentation System.
(053.01.LP2416.011)

59. 2017 NRC 059/SYS/017A4.01/3.8/1-F/RO/BANK/TS 32/053.07.LP0585.001

With the Subcooling Monitor in the 'TC' mode, which of the following is used to calculate the indicated subcooling margin?

A. The highest reading thermocouple in the core

B. The highest reading thermocouple per channel

C.✓ The average of the four thermocouples per channel

D. The average of the four highest thermocouples in each quadrant

RO Tier 2 Group 2

Source: Bank

Question History:
2005 Summer NRC RO 20

K/A:
017A4.01 In-Core Temperature Monitor System (ITM)
Ability to manually operate and/or monitor in the control room: **Actual in-core temperatures.** (Imp 3.8/4.1)

Cognitive Level:
Knowledge 1-F: The operator must recall the inputs for the Subcooled monitoring system.

Reference:
TS 32 Unit 1, Miscellaneous Equipment Checks (Monthly), Rev 25, Attachment B, Calculations, Step 2

Proposed reference to be provided to the applicants during examination:
None

Justification:

The system has four (4) thermocouples assigned to each of channel, and the average of those four (4) thermocouples is used for the temperature input for the corresponding channel.

A **INCORRECT:** Plausible if the student has a misconception on how the system functions.

B **INCORRECT:** Plausible if the student has a misconception on how the system functions, but recalls the fact that there are 2 channels of indication.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible because the use of the highest reading thermocouples is the used to determine if there is a red path.

Learning Objective:

DESCRIBE the function and/or purpose, design bases, theory of operation, and operating characteristics of the Subcooling Monitor System and its major components.
(053.07.LP0585.001)

60. 2017 NRC 060/SYS/033G2.1.32/3.8/3-SPR/RO/NEW/ROD 1.4/055.03.LP2442.009

Given the following:

- Unit 2 was shutdown for a refueling outage 137 days ago
- The outage was successfully completed and fuel assembly inventory in the Spent Fuel Pool has been unchanged since the outage
- Unit 2 is currently in MODE 1
- An event occurs which results in a loss of Spent Fuel Pool cooling
- The Crew is implementing AOP-8F, Loss of Spent Fuel Pool Cooling
- Spent Fuel Pool temperature is 86°F

Which of the following is the LONGEST expected time BEFORE reaching 200°F for the Spent Fuel Pool?
(See provided reference)

A. 13.2 hrs

B. 14.4 hrs

C.✓ 52.0 hrs

D. 56.0 hrs

RO Tier 2 Group 2

Source: New

Question History:
None

K/A:
033G2.1.32: Spent Fuel Pool Cooling System (SFPCS)
Ability to explain and apply system limits and precautions. (Imp 3.8/4.0)

Cognitive Level:
Comprehension 3-SPR: The examinee must understand the given information and use the SFP Heatup Data tables provided to determine SFP time to 200°F.

Reference:
AOP-8F, Loss of Spent Fuel Cooling, Rev 22, Step 1 Note
ROD 1.4 Unit 2, Spent Fuel Pool Heatup Data, Rev 9, Table 2/Figure 4

Proposed reference to be provided to the applicants during examination:
ROD 1.4, Spent Fuel Pool Heatup Data (entire)

Justification:

For a given temperature of 86°F, data should be taken from the 90° and 80° data columns on Table 2. The 90° column should be used giving a value of 52 hours. If other methods are use; given a heat up rate of 2.11 °F/hr (for 190 days) it will take 54 hours to reach 200°F. If interpolation of the graph is used, a value slightly higher than 54 hours will be calculated.

A **INCORRECT:** Plausible if the student uses the Table 1 not realizing this table is used before reload using the 90°F column/curve.

B **INCORRECT:** Plausible if the student uses the Table 1 not realizing this table is used before reload using the 80°F column/curve.

C **CORRECT:** See above explanation.

D **INCORRECT:** 56 – from the 80°F column of the “Core Reload” table, based on 2.11°F/hr, this would cause a temperature of 204°F.

Learning Objective:

PREDICT the effect of a loss of spent fuel pool cooling system on temperature of the spent fuel.
(055.03.LP2442.009)

61. 2017 NRC 061/SYS/041K5.05/2.6/2-RI/RO/MODIFIED/TS B 3.7.1/052.02.LP0035.006
Given the following:

- Unit 1 is operating at Rated Thermal Power
- An electrical malfunction causes a Main Generator lockout
- The Steam Dump Mode Selector switch is in AUTO

Which of the following complete this statement?
In response to the event, steam dump valves blow open, then modulate to control TAVG ___(1)___, AND RCS pressure rises and ___(2)___.

A. (1) within 4°F of TREF
(2) exceeds 110% of design pressure

B.✔ (1) at 547°F
(2) remains below 110% of design pressure

C. (1) within 4° of TREF
(2) remains below 110% of design pressure

D. (1) at 547°F
(2) exceeds 110% of design pressure

RO Tier 2 Group 2

Source: Modified

Question History:
2009 Sequoyah NRC Retake RO 62

K/A:
041K5.05 Steam Dump System (SDS) and Turbine Bypass Control
Knowledge of the operational implications of the following concepts as they apply to the SDS: **Basis for RCS design pressure limits.** (Imp 2.6/3.2)

Cognitive Level:
Comprehension 2-RI: The student must understand the initial conditions, the effect of the lockout and apply that to how the system will operate to control TAVE and RCS pressure.

Reference:
TS B 3.7.1, Main Steam Safety Valves (MSSVs), Rev 3, Background Section
883D195 Sh 17, Steam Dump Control Logic Diagram, Rev 20

Proposed reference to be provided to the applicants during examination:
None

Original Question
Given the following:
- *Unit 1 operating at 100% power when a complete loss of load event occurs.*

Which ONE of the following identifies ...
(1) how the condenser steam dump system is designed to open the steam dump valves,
and
(2) if the steam dump valves failed to open, how the reactor coolant system (RCS) design pressure limit would be impacted?

A. (1) trip open due to bistable operation
(2) would exceed 110% of the design pressure.

B. (1) trip open due to bistable operation
(2) would remain below 110% of the design pressure.

C. (1) ramp open due to controller demand
(2) would exceed 110% of the design pressure.

D. (1) ramp open due to controller demand
(2) would remain below 110% of the design pressure.

Proposed Answer: B

Justification:

For turbine trip control, the condenser steam dumps will blow open and then modulate to control TAVG at 547°F (No-load TAVG)

RCS pressure will increase due to the increase in temperature. If the condenser steam dumps fail to open on the trip, the main steam safety valves (MSSV) will operate to limit RCS pressure to less than 110% of design pressure.

From the TS B "The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 14.1.9 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting anticipated operational occurrence (AOO).

Another analysis is performed assuming no primary system pressure control, reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure."

A **INCORRECT:** Both halves are incorrect. Plausible as this describes condenser steam dumps system operation during a load rejection, and the second half is plausible because a loss of load event will result in an RCS pressure increase.

B **CORRECT:** See above explanation.

C **INCORRECT:** The first part is incorrect. Plausible as the first half describes condenser steam dumps system operation to a load rejection, and the second half is correct.

D **INCORRECT:** The second half is incorrect. Plausible as the first half is correct and the second half is plausible because a loss of load event will result in an RCS pressure increase.

Learning Objective:

Recognize and assess the response of the steam dump system to: a. Turbine Trip
(052.02.LP0035.006)

62. 2017 NRC 062/SYS/055K3.01/2.5/2-DR/RO/BANK/NUC-GFP-CMP-003/055.03.LP2439.004

Given the following:

- Unit 1 is at Rated Thermal Power
- A single set of Main Air Ejectors is in service
- 1MS-2074, Air Ejector Steam Supply control valve, fails CLOSED

Which of the following will result?
(Assume no operator action)

A.✓ Lowering megawatt output and rising hotwell temperature

B. Rising megawatt output and rising condenser hotwell level

C. Lowering megawatt output and lowering hotwell temperature

D. Rising megawatt output and rising gland seal header pressure

RO Tier 2 Group 2

Source: Bank

Question History:
2011 PBNP NRC Exam RO 63

K/A:
055K3.01 Condenser Air Removal System (CARS)
Knowledge of the effect that a loss or malfunction of the CARS will have on the following: **Main condenser.** (Imp 2.5/2.7)

Cognitive Level:
Comprehension 2-DR: The operator must understand the initial conditions of the plant, the implications of having one set of air ejectors on service, then identify the impact of the failure, determining what that failure will result in and the impact on the main generator and condenser.

Reference:
NUC-GFP-CMP-003, Heat Exchangers and Condensers, Rev 0, Section 6

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions, a degraded vacuum condition exists. This degraded vacuum will cause the megawatt output of the main generator to lower due to a reduced ΔH and the condenser is a saturated system. Being a saturated system, this changed will result in a high PSAT which will cause a higher TSAT, and therefore a higher hotwell temperature.

A **CORRECT:** See above explanation.

B **INCORRECT:** Megawatts will lower, not rise. Plausible if the student reasons that hotwell level will rise as a result of the steam system transient - rising megawatt output.

C **INCORRECT:** Degraded vacuum will cause megawatt output to lower, but hotwell temperature will rise, not lower. Plausible if the student reverses the change in absolute pressure, based on the system being a vacuum instead of a pressure.

D **INCORRECT:** Megawatts will lower not rise, gland seal header pressure may rise for more than one reason, first with the loss of the supply steam for the air ejectors, and secondly due to the degrading vacuum. Plausible if the student reverses the concept of air and non-condensable gasses and with the loss of the supply steam there is more steam for the generator.

Learning Objective:

Given access to the Site Specific Simulator or specific plant conditions, EVALUATE plant indications associated with the following events:
a. Secondary Coolant System leak
b. Feedwater system malfunction
c. Loss of Condenser Vacuum
d. Loss of Instrument Air Trip
(055.03.LP2439.004)

63. 2017 NRC 063/SYS/068A2.04/3.3/2-DI/RO/NEW/RMSASRB CI RE-223/051.04.LP0063.005
Given the following:

- 'B' Waste Distillate Tank is being discharged via RE-223, per OI-140B, Standard Radioactive Batch Liquid Release – Waste Distillate Tanks
- RE-223, Waste Distillate Release Liquid Monitor, goes into HIGH ALARM
- The PAB AO reports that Waste Distillate flow has risen since the last check and is now greater than the procedural limit of 90 gpm

Which of the following actions should be taken in response to these indications?

A. Secure P-135, Waste Distillate Pump, it failed to auto trip on the RMS alarm.

B.✓ Shut BE-LW-15, Waste Distillate Overboard Discharge Flow Control Valve; it failed to auto close on the RMS alarm.

C. Shut WL-18, Waste Condensate Overboard Discharge to SW Header Control valve; it failed to auto close on the RMS alarm.

D. Throttle Waste Distillate flow using BE-LW-15, Waste Distillate overboard Discharge Flow Control Valve, to clear the RE-223 alarm.

RO Tier 2 Group 2

Source: New

Question History:
None

K/A:
068.A2.04: Liquid Radwaste System (LRS)
Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Failure of automatic isolation.** (Imp 3.3/3.3)

Cognitive Level:
Comprehension 2-RI: The examinee must understand the given indications and recognize that the required automatic action failed to occur and then select the appropriate action to be taken.

Reference:

RMSASRB CI RE-223, Radiation Monitoring System Alarm Setpoint & Response Book Channel Information Sheet, Waste Distillate Tank Overboard Monitor, Rev 5, Step 3.2.1

OI 140, Standard Radioactive Batch Liquid Release-Waste Distillate Tanks, Rev 7, Step 5.3.7

Proposed reference to be provided to the applicants during examination:

None

Justification:

Directed per RMSASRB CI RE-223. The report of continued flow from the Waste Distillate tank confirms that the auto closure did not occur and manual actions need to be taken to secure that flow.

A **INCORRECT:** Incorrect because the pump does not have an auto trip function from RE-223. Plausible because securing the pump would stop discharge flow, which is the desired result.

B **CORRECT:** See above explanation.

C **INCORRECT:** Incorrect because RE-223 does not auto close this valve. Plausible because this is a valve can be in the discharge flow path and it has an auto close function based on RE-218.

D **INCORRECT:** Incorrect because the procedural directed action is to terminate the discharge. Plausible because reducing discharge flow may have the desired effect of clearing the RMS alarm and the examinee may believe this is a reasonable first action to address the high RMS alarm.

Learning Objective:

IDENTIFY and DESCRIBE the controls, alarms, and indications associated with the Liquid Waste Disposal System, including:

a. Location and function of component and/or system operating controls and control stations

b. Alarming locations and response to major system and component alarms

c. Plant, system, and component conditions or permissives required for operation

d. Setpoints associated with major system alarms and/or interlocks

(051.04.LP0063.005)

64. 2017 NRC 064/SYS/072A1.01/2.6/2-DR/RO/BANK/EOP-3/031.02.LP0441.008

Given the following:

- Unit 1 operating at Rated Thermal Power
- 80 gpm Letdown was established due to high reactor coolant activity from fuel failure
- A major steam leak developed downstream of the MSIVs
- The reactor tripped and Safety Injection (SI) actuated automatically
- Both MSIVs shut
- Subsequently, RCS pressure has recovered and TAVG is being maintained with Atmospheric Steam Dumps
- A tube rupture then develops in the 'A' Steam Generator

As the Control Room operators monitor their equipment, which Radiation Monitor will indicate the Steam Generator tube rupture first?

A. RE-214 PAB Exhaust Stack

B. 1RE-215 Condenser Air Ejector Radiation Monitor

C. 1RE-219 Steam Generator Blowdown Radiation Monitor

D✓ 1RE-231 'A' Main Steam Line Radiation Monitor

RO Tier 2 Group 2

Source: Bank

Question History:
None

K/A:
072A1.01 Area Radiation Monitoring (ARM) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including:
Radiation Levels. (Imp 3.4/3.6)

Cognitive Level:
Comprehension 2-DR: The operator must understand the initial conditions, then determine which detector will be the first to detect the tube rupture based on location, and time operational current condition of the core.

Reference:

EOP-3 Unit 1, Steam Generator Tube Rupture, Rev 51, Step 2
BG-EOP-3, Steam Generator Tube Rupture Background Document, Rev 36, EOP Step 2

Proposed reference to be provided to the applicants during examination:

None

Justification:

With the fuel failure, the first indication of a primary to secondary (steam generator tube) leak will be 1RE-231, ‘A’ Main Steam Line Radiation Monitor due to the activity from that fuel failure.

A **INCORRECT:** Plausible if the student believes the PAB Exhaust Stack monitor will detect steam released from the ADVs or shine based on monitor and steam pipe location.

B **INCORRECT:** Plausible because this rad monitor will alarm due to a steam generator tube leak, and the student may fail to recognize the SG is isolated from the condenser.

C **INCORRECT:** Plausible because this rad monitor will alarm due to a steam generator tube leak, and the student may fail to recognize that blowdown flow is isolated.

D **CORRECT:** See above explanation.

Learning Objective:

EVALUATE the RMS response for a Steam Generator tube leak or rupture. (031.02.LP0441.008)

65. 2017 NRC 065/SYS/079K1.01/3.0/3-PEO/RO/BANK/STPT 14.7/052.06.LP0338.004

Given the following:

- K2A, Instrument Air compressor, is selected to Constant
- K2B, Instrument Air compressor, is selected to Auto
- K3A and K3B Service Air compressors are in a normal alignment

If K2A trips and Instrument Air header pressure drops continuously, in what order will the following automatic actions occur?

1. Instrument Air Dryer Bypass opens
2. Instrument Air Low Pressure Alarm annunciates
3. Standby Instrument Air compressor starts
4. Standby Service Air compressor starts

A. 2, 1, 3, 4

B. 2, 1, 4, 3

C.✓ 3, 2, 4, 1

D. 3, 4, 2, 1

RO Tier 2 Group 2

Source: Bank

Question History:
2005 PBNP NRC RO 54

K/A:
079K1.01 Station Air System (SAS)
Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: **IAS.** (Imp 3.0/3.1)

Cognitive Level:
Comprehension 3-PEO: The operator must understand the system lineup and compressor interlocks, then take into consideration the given faults, and predict the system response.

Reference:

STPT 14.7, Secondary Systems Instrument, Service and Breathing Air Setpoint Document, Rev 18, Section 1.0

Proposed reference to be provided to the applicants during examination:

None

Justification:

Setpoints for actions are as follows:

90 PSIG - Standby IA compressor starts
89 PSIG - IA Low Pressure Alarm
85 PSIG - Standby SA compressor starts (IA header pressure)
80 PSIG - IA Dryer Bypass opens.

A **INCORRECT:** See above. Plausible based on alarms indicating non-normal lineup (auto compressor starting or bypass about to open).

B **INCORRECT:** See above. Plausible based on IA system functions occurring prior to SA system functions. SA/IA cross-connect valve come open at approximately the same pressure as the SA standby compress start, but values are not listed in the setpoint document, so are not presented here.

C **CORRECT:** See above explanation.

D **INCORRECT:** See above. Plausible based on compressors starting prior to the low pressure alarms being received and If the students does not remember the different starting pressures for the SA compressor, IA header versus SA header.

Learning Objective:

DESCRIBE the interlocks and automatic actuation setpoints associated with the Instrument and Service Air Systems, and major system components:

a. Air Compressors
b. Air Dryers and Dryer Bypass Solenoid valves
c. Compressor High Pressure Relief system
d. Receiver Relief valves
(052.06.LP0338.004)

66. 2017 NRC 066/GEN/G2.1.14/3.1/1-P/RO/MODIFIED/OP-AA-100-1000/SD 86.1 2.1.14

Which of the following is a correct example of when the plant paging system (Gaitronics) is to be used by Control Room staff in accordance with the OP-AA-100-1000, Conduct of Operations?

During normal operation use the plant paging system to announce . . .

A.✓ starting a reactor coolant pump.

B. placing a traveling water screen in hand.

C. commencing a radioactive waste discharge.

D. testing atmospheric steam dump valves per IT-90 Train A.

RO Tier 3

Source: Modified

Question History:
2011 South Texas Project NRC RO 74

K/A:
G2.1.14 Conduct of Operations
Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.
(Imp 3.1/3.1)

Cognitive Level:
Knowledge 1-P: The student must recall the requirement for plant paging system announcements during normal operations.

Reference:

OP-AA-100-1000, Conduct of Operations, Rev 20, Attachment 6, Equipment Manipulation and Status Control, Step 3.0.14.B

Proposed reference to be provided to the applicants during examination:

None

Original Question:

Which one of the following is a correct example of when the plant paging system (Public Address System) is to be used by Control Room staff in accordance with the Conduct of Operations Manual?

Use the plant paging system to …

A. announce changes in plant modes (e.g. changing from mode 3 to mode 4).

B. routinely contact plant personnel that are outside the Control Room.

C. announce Shift Manager reliefs.

D. inform plant personnel before making ANY changes in Main Turbine load.

Proposed Answer: A

Justification:

Per OP-AA-100-1000, Conduct of Operations, Attachment 6, Step 3.0.14.B, “During normal operation, the plant paging system will be used to announce the starting of major pumps and motors.”

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible because this is an infrequent operation, and stops automatic operation of the traveling screen motor.

C **INCORRECT:** Plausible because this is an infrequent operation involving radioactive processing.

D **INCORRECT:** Plausible if the student has the misconception this test will cause flow through the valve.

Learning Objective:

Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.
(SD 86.4 2.1.14)

67. 2017 NRC 067/GEN/G2.1.38/3.7/1-P/RO/BANK/OM 3.7/SD 86.1 2.1.38

While performing actions in EOP-3, Steam Generator Tube Rupture, the OS asks the Balance of Plant Operator (BOP) to “Check ruptured Steam Generator level greater than 32%.”

Which of the following BOP responses would satisfy OM 3.7, AOP and EOP Procedure Usage for Response to Plant Transients?

A. “OS, yes, ruptured Steam Generator narrow range level is 35%”

B.✓ “OS, no, ruptured Steam Generator narrow range level is 30% and rising”

C. “OS, no, ruptured Steam Generator narrow range level is increasing rapidly”

D. “OS, yes, ruptured Steam Generator narrow range level is greater than 32%”

RO Tier 3

Source: Bank

Question History:

2011 Callaway Retake NRC RO 68

K/A:

G2.1.38 Conduct of Operations

Knowledge of the station's requirements for verbal communications when implementing procedures. (Imp 3.7/3.8)

Cognitive Level:

Knowledge 1-P: The student must recall the requirement for communication when implementing AOP and EOP procedures.

Reference:

OM 3.7, AOP and EOP Procedure Usage for Response to Plant Transients, Rev 28, Step 6.9.8

Proposed reference to be provided to the applicants during examination:

None

Justification:

Per OM 3.27, AOP and EOP Procedure Usage for Response to Plant Transients, Step 6.9.8 “Operator response to the procedure readers directions shall utilize “yes” and “no” terms and provide values and trends whenever applicable”

A **INCORRECT:** Plausible as this gives a “yes/no” response, but does not have the trend.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible as this gives a ‘yes/no” and trend but no value.

D **INCORRECT:** Plausible as this is a repeat back of the original question, sbut no trend was given.

Learning Objective:

Knowledge of the station's requirements for verbal communications when implementing procedures.
(SD 86.1 2.1.38)

68. 2017 NRC 068/GEN/G2.2.3/3.8/1-I/RO/BANK/883D195 SH 21/SD 86.2 2.2.03

What condition(s) will AUTOMATICALLY close CC-LW-63 and CC-LW-64, Radwaste Component Cooling Water supply and return valves and why?

A. Unit 1 Containment Isolation signal ONLY;
To isolate non-essential loads while ensuring sufficient flow for essential loads.

B.✔ Unit 2 Containment Isolation signal ONLY;
To isolate non-seismic class 1 Radwaste piping and components.

C. Unit 1 OR Unit 2 Containment Isolation signal;
To isolate non-seismic class 1 Radwaste piping and components.

D. Unit 1 OR Unit 2 Containment Isolation signal;
To isolate non-essential loads while ensuring sufficient flow for essential loads.

RO Tier 3

Source: Bank

Question History:
None

K/A:
G2.2.3 Equipment Control
(multi-unit license) Knowledge of the design, procedural, and operational differences between units. (Imp 3.8/3.9)

Cognitive Level:
Knowledge 1-I: The student must recall the interlock for closing the valves and the purpose for that interlock.

Reference:
883D195 Sh 21, Containment Isolation/Containment Ventilation Isolation Logic Diagram, Rev 15, Note 6
DBD-02, Component Cooling Water System, Rev 16, Section 3.4 Page 3-37
Final Safety Analysis Report Rev 2017, Section 9.1

Proposed reference to be provided to the applicants during examination:
None

Justification:

883D195 Sh 21, Note 6 states, “CC-LW-63 & 64 close on Unit 2 CI only. Closure of either valve will shut down radwaste components. For valve control & radwaste shutdown logic see S&W Dwg. 12137-LSK-9B”
The DBD-02 states “These normally open valves shall isolate via remote manual operator actions from the control room to isolate non-seismic CC radwaste piping following a seismic event.”
The FSAR states “The auto closure of LW-63 & 64 enhances CC system integrity. This feature is not credited for the mitigation of any analyzed accident. Manual action, … is the analyzed method of CC system restoration, even under DBA conditions.”

A **INCORRECT:** Both are incorrect. Plausible if the student has a misconception of which unit causes the signal to generate (but for Unit 2 only) and this is the reason for the manual action, not the automatic action.

B **CORRECT:** See above explanation.

C **INCORRECT:** The second half is correct. Plausible if the student reasons that either unit CI should close the valves since it is a common system.

D **INCORRECT:** Both are incorrect. Plausible if the student reasons that either unit CI should close the valves since it is a common system, and this is the reason for the manual action, not the automatic action.

Learning Objective:

(multi-unit license) Knowledge of the design, procedural, and operational differences between units.
(SD 86.2 2.2.03)

69. 2017 NRC 069/GEN/G2.2.14/3.9/1-P/RO/NEW/OM 3.41/SD 86.2 2.2.14
Given the following:

- A piece of equipment is placed in an OFF-NORMAL position and is NOT procedurally controlled
- This equipment does not affect any LCOs or TLCOs
- An action request(AR)/work request(WR) to document the condition has been written

In accordance with OM 3.41, System Status Control, what additional actions are required to track the position of the shut gauge isolation?

A✓ Enter the component on the Abnormal Alignment List AND attach an Information Tag as soon as practicable.

B. Enter the component information in the Station Log AND ensure an item is included on the appropriate Turnover Notes.

C. No additional actions are required; the AR/WR is sufficient to ensure the component is tracked via the work control process.

D. Annotate the system checklist for the component out of the normal position AND maintain that checklist in the Procedure-in-Progress file.

RO Tier 3

Source: New

Question History:
None

K/A:
G2.2.14 Equipment Control
Knowledge of the process for controlling equipment configuration or status. (Imp 3.9/4.3)

Cognitive Level:
Knowledge 1-P: The operator must recall the requirements of the procedure.

Reference:
OM 3.41, System Status Control, Rev 9, Attachment B Step 4.2.3

Proposed reference to be provided to the applicants during examination:
None

Justification:
From OM-3.41

"4.2.3 When equipment is placed in an Off-normal positon and is NOT procedurally controlled, ENSURE the following for any equipment and any affected instrumentation, switches, etc.:

- An action request (AR) with a 90 day due date, and a work request (WR) are initiated to track and correct the condition as needed.

The following entered into the **Abnormal Alignment List:**

- Reason for alignment
- AR number
- Original abnormal alignment date
- EC number (if applicable)

- 10CFR50.59 tracking number (AR number) is placed in the 50.59 tracking AR column of the Abnormal Alignment List
- All applicable information fields are filled in on the Information Tag

Information Tag is affixed to the identified equipment as soon as practicable.

Purple Abnormal Alignment magnetic arrows should be used on the Main Control Boards in lieu of information tags."

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible because these are reasonable actions to take for this situation. Station log entries are a commonly required action.

D **INCORRECT:** Plausible if the examinee believes that the work control process is appropriate to maintain component status.

C **INCORRECT:** Plausible because maintaining procedures in the procedure-in –progress file is an appropriate status control action for when a procedure is suspended.

Learning Objective:
Knowledge of the process for controlling equipment configuration or status.
(SD 86.2 2.2.14)

70. 2017 NRC 070/GEN/G2.2.15/3.9/1-P/RO/BANK/OM 3.41/SD 86.2 2.2.15

Given the following:

- You are an RO on relief crew
- You are verifying a valve lineup per the applicable system checklist and drawing
- During your verification of the checklist you discover that a valve shown on the system drawing is NOT listed on the applicable checklist

Which of the following describes the required actions for this plant configuration situation?

Notify the relief crew OS, then initiate . . .

A. a Condition Report (CR) and pen and ink the checklist to continue.

B. an information tag to position the valve per the drawing, no additional action is required.

C. a Work Request (WR) and an Engineering Change Request (ECR) to update the checklist.

D.✓ a Condition Report (CR) and a Procedure Change Request (PCR) to update the checklist.

RO Tier 3

Source: Bank

Question History:
2011 PBNP NRC RO 68

K/A:
G2.2.15 Equipment Control
Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. (Imp 3.9/4.3)

Cognitive Level:
Knowledge 1-P: The student must recall the actions required when an error is discovered during the verification of a valve lineup.

Reference:

OM 3.41, System Status Control, Rev 9, Attachment A, Handling of Lineup Discrepancies, Rev 9

AD-AA-100-1004, Preparation, Revision, Review And Approval Of Site-Specific Procedures, Rev 22, Section 4.1.2.D

Proposed reference to be provided to the applicants during examination:

None

Justification:

A condition report (CR) as well as a Procedure Change Request (PCR) is required prior to continuing.

A **INCORRECT:** The first part is correct the second is not. Plausible as this is the action taken for a typographical type error, not a technical one.

B **INCORRECT:** Both parts are incorrect. Plausible the valve may be tagged for various reasons, including this one, but an update to the checklist still needs to be performed.

C **INCORRECT:** Both parts are incorrect. Plausible as the first part is used to address equipment issues and the second it to correct drawings.

D **CORRECT:** See above explanation.

Learning Objective:

Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. (SD 86.2 2.2.15)

71. 2017 NRC 071/GEN/G2.3.7/2.9/1-P/RO/BANK/NP 4.2.27/SD 86.3 2.3.7
Given the following:

- A non-licensed operator has just completed an **emergency entrance** into the RCA through the RCA checkpoint, including use of a SRD located on top of the RCA Fire Brigade locker
- The initial SRD total reading was **12 mrem**
- The SRD was reading **18 mrem** when the RCA was exited

Which of the following describes the Operator's action concerning the SRD and dose received on exiting the RCA?

A.✓ Report 6 mrem dose to the Shift Manager and return the SRD to the RP Group.

B. Report 6 mrem dose to the RP Group and return the SRD to the top of the RCA Fire Brigade locker.

C. Report 18 mrem dose to the RP Group and return the SRD to the RP Group.

D. Report 18 mrem dose to the Shift Manager and return the SRD to the top of the RCA Fire Brigade locker.

RO Tier 3

Source: Bank

Question History:
2009 PBNP NRC RO 72

K/A:
G2.3.7 Radiation Control
Ability to comply with radiation work permit requirements during normal or abnormal conditions. (Imp 3.5/3.6)

Cognitive Level:
Knowledge 1-P: The student must recall the knowledge of the dosimetry requirements associated with an emergency entry into the RCA, and who the dose received is reported to upon exit as required by the RWP/Procedure.

Reference:
NP 4.2.27, Personnel Exposure Monitoring Device Minimum Requirements and General Use, Rev 24, Step 4.5.2

Proposed reference to be provided to the applicants during examination:
None

Justification:

Per procedure the dose of 6 mrem is reported to the Shift Manager, the SRD returned to the RP Group

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible if the student has a misconception of who the dose is reported to and where to turn in the SRD, as this is where the SRD was taken from.

C **INCORRECT:** Plausible if the student has a misconception of who to report the dose to and what dose should be reported.

D **INCORRECT:** Plausible if the student has a misconception of what dose to report and where to turn in the SRD to.

Learning Objective:

Ability to comply with radiation work permit requirements during normal or abnormal conditions.
(SD 86.3 2.3.7)

72. 2017 NRC 072/GEN/G2.3.11/3.8/1-I/RO/BANK/EOP-3/031.02.LP0441.002

Given the following:

- Unit 1 was initially at Rated Thermal Power
- A Steam Generator Tube Rupture occurred in the ‘A’ SG
- EOP-3, Steam Generator Tube Rupture, is in progress
- The crew is taking action to isolate the ‘A’ SG

The ‘A’ SG Atmospheric Dump Valve (ADV) controller is set to ___(1)___
AND
the ADV is adjusted to this value to ___(2)___.

A. (1) 1005 psig
(2) minimize radiation release via the ADV and prevent lifting SG safety valves

B. (1) 1005 psig
(2) promote thermal stratification to prevent an uncontrolled depressurization during the cooldown

C.✓ (1) 1050 psig
(2) minimize radiation release via the ADV and prevent lifting SG safety valves

D. (1) 1050 psig
(2) promote thermal stratification to prevent an uncontrolled depressurization during the cooldown

RO Tier 3

Source: Bank

Question History:
2012 Vogtle NRC Exam RO 65

K/A:
G2.3.11 Radiation Control
Ability to control radiation releases. (Imp 3.8/4.3)

Cognitive Level:
Knowledge 1-I: The examinee must recall the EOP step setpoint for ruptured SG ADV and the basis of this action.

Reference:
EOP-3 Unit 1, Steam Generator Tube Rupture, Rev 51, Step 3 RNO.a or b.4)
BG-EOP-3, Steam Generator Tube Rupture, Rev 36, EOP Step 3

Proposed reference to be provided to the applicants during examination:
None

Justification:
Step 3.a or 3.b of EOP-3 will "Ensure 1HC-468/478, "A/B" ADV Controller – Set to 1050 psig"
EOP Step 3 of the background states the purpose is "to isolate flow from the ruptured steam generator to minimize radiological releases." And "The atmospheric dump valve on the ruptured steam generator should remain available to limit steam generator pressure unless it fails open. This will minimize any challenges to the code safety valve.

A **INCORRECT:** The first part is incorrect. Plausible because the ADVs will be set to 1005 psig in EOP-0, and minimizing release is correct.

B **INCORRECT:** Both halves are incorrect. Plausible because the ADVs will be set to 1005 psig in EOP-0, and preventing uncontrolled depressurization of the rupture SG is a goal of the EOP.

C **CORRECT**: See above explanation.

D **INCORRECT:** The second half is incorrect. Plausible because the ADV setpoint is correct and preventing uncontrolled depressurization of the ruptured SG is a goal of the EOP.

Learning Objective:
LIST the major actions accomplished by the following procedures:
a. EOP-2
b. EOP-3
c. EOP-3.1
d. EOP-3.2
e. EOP-3.3
(031.02.LP0441.002)

73. 2017 NRC 073/GEN/G2.3.15/2.9/2-DI/RO/NEW/TS 3.4.15/SD 86.3 2.3.15

Given the following:

- Unit 1 is operating at Rated Thermal Power
- At 0600, 1RE-127, Containment High Range Radiation monitor was declared inoperable
- At 0700, 1RE-212, Containment Noble Gas monitor was declared inoperable
- At 0800, 1RE-211, Containment Air Particulate monitor was declared inoperable
- At 0900, annunciator C01 C 3-6, UNIT 1 SUMP A LEVEL HIGH alarm was declared inoperable

Which of the following identifies the EARLIEST time LCO 3.4.15, RCS Leakage Detection Instrumentation, was NOT met

A. 0600

B. 0700

C.✓ 0800

D. 0900

RO Tier 3

Source: New

Question History:
None

K/A:
G2.3.15 Radiation Control
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (Imp 2.9/3.1)

Cognitive Level:
Comprehension 2-RI: The student must recall the requirements of the LCO, and determine which of the detectors are required, and how of those detectors there are, THEN based on when they are declared inoperable, determine when the LCO is not met.

Reference:
TS 3.4.15 RCS Leakage Detection Instrumentation, Rev 2, LCO Section
TS B 3.4.15 Background for RCS Leakage Detection Instrumentation, Rev 2, LCO Section

Proposed reference to be provided to the applicants during examination:
None

Justification:
The LCO is met by diverse means of a containment sump level alarm, in combination with a gaseous OR particulate radioactivity monitor. Based on the initial conditions, the loss of 1RE-127 will have no effect on this LCO. The loss of 1RE-212 will affect the LCO, but it will still be MET as this is one of the two available monitors, and the other monitor is still operable. The loss of 1RE-211 will mean that both available monitors are inoperable; therefore, the LCO is not met at this time.

A **INCORRECT:** Plausible if the student has the misconception that 1RE-127 is one of the monitors required by the LCO.

B **INCORRECT:** Plausible if the student knows that 1RE-212 is one of the monitors required by the LCO, has the misconception and that both have to be operable in order to meet the LCO.

C **CORRECT:** See above explanation.

D **INCORRECT:** Plausible if the student has the misconception that radiation monitor portion of the LCO is still met, but now the sump level indications portion is not met, and therefore the LCO is no longer met.

Learning Objective:
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
(SD 86.3 2.3.15)

Given specific plant conditions, ASSESS and APPLY Technical Specification requirements as appropriate.
(057.02.LP3339.002)

74. 2017 NRC 074/GEN/G2.4.14/3.8/3-SPR/RO/NEW/OM 3.7/031.01.LP0158.009

Given the following:

- A Loss Of Coolant Accident is in progress on Unit 1
- The crew is implementing EOP-1, Loss of Reactor or Secondary Coolant and preparing to secure the last spray pump per Step 12.h
- Containment pressure is 23 psig

h. Check if last spray pump can be stopped: 1) Check containment pressure - LESS THAN 20 psig	h. OBSERVE CAUTION PRIOR TO STEP 13 and go to Step 13. WHEN spray termination criteria are met, THEN do Step 12.i.
i. Stop containment spray pump and place in standby:	

- Prior to meeting the conditions of Step 12.h, an ORANGE condition requiring entry into CSP-P.1, Response to Imminent Pressurized Thermal Shock Condition, occurs
- While implementing CSP-P.1, Containment pressure decreases to below 20 psig

What action, if any, should the crew take and what is the basis?

A✓ DO NOT perform Step 12.i; EOP-1 actions are superseded by the ORANGE CSP.

B. Immediately perform Step 12.i; Step 12 is a continuous action step and still in effect.

C. Immediately perform Step 12.i; conditions described by the WHEN/THEN statement are satisfied.

D. Step 12.i may be performed if the Unit Supervisor determines there is no conflict with the procedure in effect.

RO Tier 3

Source: New

Question History:
None

K/A:
G2.4.14 Emergency Procedures /Plan
Knowledge of general guidelines for EOP usage. (Imp 3.8/4.5)

Cognitive Level:
Application 3-SPR: The examinee must recall the EOP rules of use and APPLY those rules to the given scenario and procedure step.

Reference:
OM 3.7, AOP and EOP Procedure Usage for Response to Plant Transients, Rev 28, Step 6.15.2 and 6.15.3

Proposed reference to be provided to the applicants during examination:
None

Justification:
OM 3.7 Step 6.15.3 prohibits the performance of EOP/ECA actions if a RED or ORANGE path CSP is in effect.

A **CORRECT:** See above explanation.

B **INCORRECT**: Plausible because CA steps remain in effect "until superceded by alternate guidance or stated to be inapplicable." (6.15.3)

C **INCORRECT:** Plausible because this would be the correct action if EOP-1 were still in effect. (6.15.2)

D **INCORRECT:** Plausible because this would be an appropriate action for a WHEN/THEN condition from a previously exited procedure. (See 6.15.2)

Learning Objective:
Given the appropriate emergency procedures, demonstrate the ability to implement the standard rules of usage associated with the EOPs. (031.01.LP0158.009)

75. 2017 NRC 075/GEN/G2.4.45/4.1/3-SPK/RO/BANK/ARB C01 B 2-9/SD 86.4 2.4.45
Given the following:

- A Loss of Coolant Accident has occured
- 1P-53, Motor Driven AFW pump, is running, supplying water to the Steam Generators
- Both Steam Generator levels are 40% and RISING SLOWLY
- The Crew has returned to EOP-1, Loss of Reactor or Secondary Coolant, from EOP-1.3, Transfer to Containment Sump Recirculation – Low Head Injection, following sump recirc preparation steps
- During the performance of EOP-1.3 alignment, the crew noted that Containment radiation levels were beginning to rise
- RWST level was at 40% and lowering when the crew returned to EOP-1
- Offsite power is available
- Containment pressure is 4 psig and LOWERING SLOWLY

If all the alarms listed below were received CONCURRENTLY, which of the alarms would be given the highest priority?

A. C01 A 4-9, AUX FEED PUMP SUCTION PRESSURE LOW

B.✓ C01 B 2-9, 1T-13 RWST LEVEL LOW

C. C02 E 2-2, G-03 EMERGENCY DIESEL

D. 1C20 D 1-9, UNIT 1 CONTAINMENT HIGH RANGE RADIATION HIGH

RO Tier 3

Source: Bank

Question History:
2005 PBNP NRC RO 75

K/A:
G2.4.45 Emergency Procedures /Plan
Ability to prioritize and interpret the significance of each annunciator or alarm. (Imp 4.1/4.3)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, and determine the plant conditions which would allow the crew transitioned back to EOP-1, after determining what would have caused the annunciators to alarm, and then prioritize which one of the causes has the highest priority.

Reference:

ARB C01 B 2-9, 1T-13 RWST LEVEL LOW, Rev 5
EOP-1.3 Unit 1, Transfer to Containment Sump Recirculation – Low Head Injection, Rev 55, Step 22
STPT 11.1, Safety Injection System General Instrumentation Channels Setpoint Document, Rev 20, Section 1

Proposed reference to be provided to the applicants during examination:

None

Justification:

Given the initial conditions, the line up for sump recirc has been initiated per EOP-1.3, the crew returned to EOP-1 due to having no further actions to take in EOP-1.3 until RWST level reaches the appropriate level. The annunciator indicated the RWST level has reached 34%, and the crew shall immediately return to EOP-1.3 and commence step 23.

A **INCORRECT:** Plausible as the aux feed pump suction pressure low indicates there is an issue with the suction of the AFW pumps, either the motor or turbine driven AFW pump. While there is no status of the amount of AFW flow, or which pump is supplying, the SG level is 40% and the containment is not adverse, therefore there is no required AFW flow.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible because safeguard busses are needed to mitigate the LOCA, but currently both safeguard busses are being supplied by off-site power. This annunciator is caused by an issue with the EDG such as high generator temperature, low water level, trip/lockout, etc., This is an important annunciator, but does not out weight the need for sump recirc.

D **INCORRECT:** Plausible due to the LOCA, this can be an issue, and should be investigated to determine the level of radiation in containment, but should not take precedent before the establishment of sump recirc.

Learning Objective:

EVALUATE the RMS response for a Steam Generator tube leak or rupture. (031.02.LP0441.008)

WRITTEN / ORAL / ONLINE EXAMINATION KEY COVER SHEET

Examination Number/Title: SRO PORTION ONLY						
Training Program: PBN LOI Initial License Training Program						
Course/Lesson Plan Number(s): SRO PORTION ONLY						
Total Points Possible: 25.00	**PASS CRITERIA: ≥ 80%**		**Exam Time:**			
	Yes	**No**		**Yes**	**No**	
This is an alternate examination; verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☐	This is a remediation exam. Verified the questions are different from the failed exam by at least the following criteria listed below: • 70% for Maintenance/Technical • 90% for Operations training programs	☐	☐	
This is an initial training examination; verified at least 30% of the questions are different from previous administration of the same exam. This requirement is NA if ≥ 2 years since previous administration of the exam.	☐	☐	This is a LOCT annual operating exam or biennial comprehensive remedial exam, verified the questions are 100% different from the failed exam.	☐	☐	
This is a non-randomly generated exam from an electronic exam bank, printed out or administered online. Verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified ≥ 50% difference.	☐	☐	This is a randomly generated exam from an electronic exam bank, printed out or administered online. Verified the exam bank has 3 questions per objective if one test item on exam for the objective. If 2 or more test items on exam for an objective, then 6 questions are in bank.	☐	☐	

NOTE:
• See TR-AA-230-1003, SAT Development, for exam development and review guidelines.
• NRC exams may require additional information. Refer to fleet and site specific procedures.
			Key should contain the following:
			• Learning Objective Number
			• Test Item
			  • Question or Statement
			  • All possible answers
			  • Correct Answer Indicated
			  • Point Value
			  • References (if applicable)			

EXAMINATION REVIEW AND APPROVAL:	
Developed by:	**Date:**
Instructional Review of Written Exam (Qualified Instructor):	**Date:**
Technical Review (SME):	**Date:**
Approved by Training Supervisor:	**Date:**
Approved by Training Program Owner (or line designee):	**Date:**

Indicate in the following table if any changes are made to the exam after approval:

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR# (if applicable)	PREPARER SUPERVISOR	DATE DATE

#	ID	Points	Type	Answers 0
1	2017 NRC 76	1.00	MCS	B
2	2017 NRC 77	1.00	MCS	B
3	2017 NRC 78	1.00	MCS	C
4	2017 NRC 79	1.00	MCS	B
5	2017 NRC 80	1.00	MCS	D
6	2017 NRC 81	1.00	MCS	A
7	2017 NRC 82	1.00	MCS	C
8	2017 NRC 83	1.00	MCS	A
9	2017 NRC 84	1.00	MCS	A
10	2017 NRC 85	1.00	MCS	B
11	2017 NRC 86	1.00	MCS	D
12	2017 NRC 87	1.00	MCS	A
13	2017 NRC 88	1.00	MCS	C
14	2017 NRC 89	1.00	MCS	B
15	2017 NRC 90	1.00	MCS	D
16	2017 NRC 91	1.00	MCS	B
17	2017 NRC 92	1.00	MCS	B
18	2017 NRC 93	1.00	MCS	D
19	2017 NRC 94	1.00	MCS	A
20	2017 NRC 95	1.00	MCS	D
21	2017 NRC 96	1.00	MCS	B
22	2017 NRC 97	1.00	MCS	B
23	2017 NRC 98	1.00	MCS	D
24	2017 NRC 99	1.00	MCS	C
25	2017 NRC 100	1.00	MCS	C

1. 2017 NRC 076/APE/022AA2.03/3.6/3-SPR/SRO/BANK/TRM 3.5.1/057.02.LP3410.003
Given the following:

- Unit 1 is in a refueling outage with half of the core unloaded to the Spent Fuel Pool
- Unit 2 is at Rated Thermal Power
- Unit 1 RWST is drained for maintenance
- Unit 2 RWST is aligned normally

- The Boric Acid Storage Tanks (BAST) contain a 3.75% boric acid solution
- The Unit 1 Reactor Operator reports the following:
  - 'A' BAST level is 48% with a temperature of 73°F
  - 'B' BAST level is 92% with a temperature of 71°F
  - 'C' BAST level is 85% with a temperature of 68°F
- The PAB operator reports the following:
  - BS-331, 'B' BAST outlet isolation valve has been found **SHUT** and not able to be opened

What Technical Requirement Manual actions, if any, should the SRO direct?
(See provided references)

A. No actions needed. TLCO 3.5.1 is **met** for **both** units.

B✔ Declare TLCO 3.5.1 **NOT** met for Unit 1 and enter only TRMAC 3.5.1.A for Unit 1.

C. Declare TLCO 3.5.1 **NOT** met for Unit 2 and enter only TRMAC 3.5.1.C for Unit 2.

D. Declare TLCO 3.5.1 **NOT** met for **both** units and enter TRMAC 3.5.1.A for Unit 1 and TRMAC 3.5.1.C for Unit 2.

SRO Tier 1 Group 1

Source: Bank

Question History:
2009 PBNP NRC SRO 11

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

K/A:
022AA2.03 Loss of Reactor Coolant Makeup
Ability to determine and interpret the following as they apply to | the Loss of Reactor Coolant Makeup: **Failures of flow control valve or controller.**
(Imp 3.1/3.6)

Cognitive Level:
Comprehension 3-SPR: The operator must understand the initial conditions and sequence of events, then using the provided references determine what the impact will be on shutdown safety and technical specifications.

References:
TRM 3.5.1 Chemical and Volume Control System, Rev 11, Table 3.5.1-1, TLCO and Applicability,
TLB 5, Boric Acid Storage Tanks ID W 685-J-105 T6 A/B/C, Rev 10

Proposed reference to be provided to the applicants during examination:
TRM 3.5.1, Pages 1 - 4
TLB 5, Pages 1 and 2 (ALL)

Justification:

Use TRM and TLB to calculate approximately 2214 gal of acid available to Unit 1 from the 'A' BAST. With Unit 1 being in MODE 6, only one flow path for boric acid injection to the reactor is required. However, given the initial conditions, ‘B’ BAST is isolated, and ‘C’ BAST is too cold, so only ‘A’ BAST is available, thus the TLCO is NOT met. The TLCO is MET for Unit 2 due to the RWST being available with multiple flow paths.

A **INCORRECT:** Plausible if the student utilized the tank calculation and does not subtract the 462.79 gal (i.e., 48% x 55.74 = 2675 gal) and then the amount will meet the requirement of note (5) for TRM Table 3.5.1-1. Note (5) states: If the unit has been borated to meet the refueling shutdown margin requirement… the minimum combined volume for that unit is 2,670 gallons.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible if the student misapplies the requirements for BASTs to Unit 2. Flow paths from the BASTs is not required from Unit 2 due to the RWST being available.

D **INCORRECT:** Plausible if the student correctly determines the TLCO requirement is not met for Unit 1 and fails to credit the Unit 2 RWST but instead applies the information that the ‘B’ BAST is isolated, and ‘C’ BAST being too cold.

Learning Objective:

Given specific plant conditions, ASSESS and APPLY Technical Specification Technical Requirements Manual requirements as appropriate. (057.02.LP3410.003)

2. 2017 NRC 077/APE/026AG2.4.8/4.5/3-SPK/SRO/NEW/AOP-9B/055.03.LP2444.001

Given the following:

- Unit 1 is operating at 28% of Rated Thermal Power
- A Component Cooling Malfunction occurs
- The crew is responding per AOP-9B, Component Cooling System Malfunction
- 1TI-612, 1P-1A RCP CC Outlet Temperature is 102°F and RISING SLOWLY
- 1TI-608, 1P-1B RCP CC Outlet Temperature is 100°F and STABLE
- 1CC-815, 1TI-12 CC Surge Tank Emer MU MOV is OPEN
- 1LI-618B, CC Surge Tank Level is 44% and LOWERING SLOWLY
- The following annunciators are LIT
    - 1C03 1D 1-4, 1P-1A RCP COOLING WATER FLOW LOW
    - 1C03 1D 3-6, 1T-12 CC SURGE TANK LEVEL HIGH OR LOW
    - C01 B 1-3, UNIT 1 CONTAINMENT SUMP A LEVEL HIGH

Which of the following identifies the correct set of sequences the OS1 will direct?

A. 1. Trip the Reactor
   2. Trip ‘A’ RCP
   3. Branch to EOP-0, Reactor Trip or Safety Injection and stabilize the plant
   4. Shut CCW cooling inlet and outlet MOVs to RCP ‘A’

B✓ 1. Branch to AOP-1B, Reactor Coolant Pump Malfunction
   2. Trip the Reactor and stabilize the plant with EOP-0, Reactor Trip or Safety Injection
   3. Trip ‘A’ RCP
   4. Shut CCW cooling inlet and outlet MOVs to RCP ‘A’

C. 1. Trip the Reactor and stabilize the plant with EOP-0, Reactor Trip or Safety Injection
   2. Branch to AOP-1B, Reactor Coolant Pump Malfunction
   3. Trip ‘A’ **AND** ‘B’ RCP
   4. If CCW Surge tank level continues to lower shut thermal barrier cooling return AOVs for 'A' **AND** 'B' RCPs

D. 1. Transition to AOP-1B, Reactor Coolant Pump Malfunction
   2. Trip the Reactor
   3. Trip ‘A’ **AND** ‘B’ RCP
   4. Branch to EOP-0, Reactor Trip or Safety Injection and stabilize the plant

SRO Tier 1 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
026AG2.4.08 Loss of Component Cooling Water (CCW)
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (Imp 3.8/4.5)

Cognitive Level:
Comprehension 3-SPK: The student must understand the initial condition, using system knowledge determine the implications of the leak and the location, apply that information to the initial condition, and determine the sequence of events required to mitigate the situation per the procedures.

Reference:
AOP-9B Unit 1, Component Cooling System Malfunction, Rev 24, Step A9
AOP-1B Unit 1, Reactor Coolant Pump Malfunction, Rev 24, Foldout Page Item 2
ARP 1C03 1D 1-4, 1P1A RCP COOLING WATER FLOW LOW, Rev 3, Section 3.2

Proposed reference to be provided to the applicants during examination:
None

Justification:

Based on indications the ARP for RCP low flow will send the crew to AOP-9B. AOP-9B will deal with the surge tank level as stated in the question stem and also attempt to diagnose the leak. Based on the RCP low flow, and surge tank level lowering, there is a leak in the portion of CCW that cools the RCP. AOP-9B, will have you branch to AOP-1B to secure the pump, and then return to isolate the individual heat exchanger using manual valves in the containment. Once the crew branches to AOP-1B, they will meet the Foldout Page Item 2 which will require a plant trip, stabilization using the EOPs then securing the RCP and shutting the MOVs to isolate CCW to the affected RCP.

A **INCORRECT:** Transition to EOP-0 is done prior to securing the RCP to stabilize the plant. Plausible if the student places importance on immediately tripping the RCP.

B **CORRECT:** See above explanation

C **INCORRECT:** The ’B’ RCP is not required to be tripped, this will complicate the recovery by forcing the crew to utilize natural circulation to cool down the plant. Also shutting just the CC-761A/B valves will not stop the leak from the CCW system. Plausible if the student determines a rising temperature in the ‘B’ RCP requires it to be tripped, and misdiagnoses the leak. Branching to AOP-1B would be correct at this point.

D **INCORRECT:** Transition to EOP-0 is done prior to securing the RCP to stabilize the plant. Plausible if the student places importance on immediately tripping the RCPs.

Learning Objective:

Given appropriate system/equipment conditions and indications, DETERMINE when a loss of the following systems has occurred based on Control Room alarms:

a. Component Cooling Water (CCW)
b. Service Water (SW)
c. Circulating Water (CW)

(055.03.LP2444.001)

3. 2017 NRC 078/EPE/029EA2.09/4.5/3-SPK/SRO/MODIFIED/CSP-S.1/043.03.LP1996.013
Given the following:

- Unit 1 is operating at 25% power when an earthquake is felt onsite.
- Main condenser vacuum begins rapidly degrading.
- The crew enters AOP-5A, Loss of Condenser Vacuum.
- While preparing to reduce power, the main turbine trips on low vacuum.
- The STA reports that CST level is 3.5 feet and LOWERING

Which of the following actions are required?
(AOP-5A, Loss of Condenser Vacuum)
(AOP-23, Establishing Alternate AFW Suction)
(AOP-25, Turbine Trip Without Reactor Trip)
(EOP-0, Reactor Trip or Safety Injection)
(CSP-S.1 Response to Nuclear Power Generation/ATWS)

A. Automatic reactor trip from turbine trip is not required at this power level.
The Operating Supervisor should direct a manual reactor trip per AOP-5A, then enter EOP-0. Complete the immediate actions then perform AOP-23 in parallel.

B. Automatic reactor trip from turbine trip is not required at this power level.
The Operating Supervisor should respond to the turbine trip with AOP-25, and perform a controlled shutdown. AOP-23 should be performed in parallel.

C.✓ An automatic reactor trip should have occurred.
The Operating Supervisor should direct a manual reactor trip per EOP-0. If not successful, enter CSP-S.1. Complete the immediate actions and then perform AOP-23 in parallel with CSP-S.1 actions.

D. An automatic reactor trip should have occurred.
The Operating Supervisor should direct a manual reactor trip per EOP-0. If not successful, enter CSP-S.1. Complete the immediate actions and then transition to AOP-23 based on the Foldout Page, RETURN to CSP-S.1 when AOP-23 actions are completed.

SRO Tier 1 Group 1

Source: Modified

Question History:
2011 PBNP SRO 78

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
029EA2.09 Anticipated Transient Without Scram (ATWS)
Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip. (Imp 4.4/4.5)

Cognitive Level:
Comprehension 3-SPK: The student must understand the initial condition, use procedure knowledge to determine if foldout page criteria is met, and rules of usage for determining the correct action.

Reference:
883D195 Sh 3, Turbine Trip Signals Logic Diagram, Rev 35
883D195 Sh 12, Nuclear Instr. Permissives and Blocks Logic Diagram, Rev 10
CSP-S.1 UNIT 1, Response to Nuclear Power Generation/ATWS, Rev 40, Foldout Page Item 2
OM 3.7, AOP and EOP Procedure Usage for Response to Plant Transients, Rev 28, Sections 6.16 and 6.17.3

Proposed reference to be provided to the applicants during examination:
None

Original Question:

Given the following conditions:

- *Unit 1 experienced an ATWS*
- *The turbine failed to trip when the turbine trip pushbutton was depressed while performing CSP-S.1, 'Response to Nuclear Power Generation/ATWS'.*
- *While the Control Operator is addressing the failure of the turbine to trip, the STA reports that CST level is 8 feet and lowering.*
  (AOP-23 UNIT 1, 'ESTABLISHING ALTERNATE AFW SUCTION')

Which of the following actions are required?

A. IMMEDIATELY transition *to AOP-23 from the Foldout Page, THEN return to CSP-S.1 when all AOP-23 actions are completed.*

B*. Continue performing CSP-S.1 actions until directed to return to procedure in effect, THEN* ***branch*** *to AOP-23 from the Foldout page and perform in parallel with procedure in effect.*

C *Complete the remaining actions to trip the turbine, THEN* ***branch*** *to AOP-23 from the Foldout Page and perform in parallel with CSP-S.1.*

D*. Complete the remaining actions to trip the turbine, THEN* ***transition*** *to AOP-23 from the Foldout Page, returning to CSP-S.1 when all AOP-23 actions are completed.*

Answer: C

Justification:

This question requires the candidate to determine whether a reactor trip should have occurred following a turbine trip and then further tests the SRO's direction of parallel AOP procedure use while in the ATWS CSP. With the loss of condenser vacuum, interlock P-9 is not met which removes the block of the Reactor Trip from Turbine trip. The loss of Condensate Storage Tank level is a serious concern, and the SRO needs to follow EOP/AOP rules of usage to address that without impeding the actions required to address the ATWS condition. Based on procedure use rules, foldout page items are dealt with by branching, *after* immediate actions are completed and verified by the SRO.

A **INCORRECT:** Automatic reactor trip is required. This is plausible if the student forgets the condenser vacuum contribution to P-9 logic. The SRO procedure flowpath is otherwise correct if the reactor manual trip is successful.

B **INCORRECT:** Automatic reactor trip is required. This is plausible if the student forgets the condenser vacuum contribution to P-9 logic. The second part is not otherwise correct because AOP-5A would direct a manual reactor trip and AOP-25 would not be used in that case (although it is normally used for Turbine Trip without Reactor Trip).

C **CORRECT:** See above explanation.

D **INCORRECT:** The first part is correct, a reactor trip should have occurred. The CST level is important to address, but the AOP-23 actions have to be done in parallel with CSP actions, and not in series.

Learning Objective:

IMPLEMENT the CSPs to respond to plant conditions where the Subcriticality Status Tree is not satisfied.
(043.03.LP1996.013)

4. 2017 NRC 079/APE/054G2.4.4/4.7/3-SPK/SRO/BANK/AOP-2B/PB-052-05-LP0128-005
Given the following:

- Unit 1 is operating at 70% of Rated Thermal Power
- The following annunciators are received:
  - 1C03 1E2 1-7, FEEDWATER PUMPS TRIP
  - 1C03 1E2 1-2, STEAM GENERATOR A LEVEL SETPOINT DEVIATION/TROUBLE
  - 1C03 1E2 1-5, STEAM GENERATOR B LEVEL SETPOINT DEVIATION/TROUBLE
  - 1C03 1E2 3-7,LOSS OF FW TURBINE TRIP CHANNEL ALERT

- Both Steam Generator levels are 61% and lowering
- 1P-28A, Main Feed Pump control switch Red Light **OFF** Green Light **ON**
- 1P-28B, Main Feed Pump control switch Red Light **ON** Green Light **OFF**

Which of the following Operator Actions should the OS1 direct the crew to take?

A. Manually trip the Reactor and go to EOP-0, Reactor Trip or Safety Injection as directed by AOP-2B, Feedwater System Malfunction

B.✔ Reduce power to less than 60% per AOP-17A, Rapid Power Reduction while addressing MFP trip with AOP-2B, Feedwater System Malfunction

C. Reduce Turbine Load to less than 50% using AOP-17A, Rapid Power Reduction and manually trip the Main Turbine per AOP-25, Turbine Trip without reactor trip

D. Ensure Feed Water Regulating Valves are in automatic 3-element mode and monitor that levels are restored to normal per ARP 1CO3 1E2 1-2 (1-5) Steam Generator A (B) Level Setpoint Deviation/Trouble

SRO Tier 1 Group 1

Source: Bank

Question History:
2011 Callaway NRC SRO 79

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
054G2.4.4 Loss of Main Feedwater (MFW)
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (Imp 4.5/4.7)

Cognitive Level:
Comprehension 3-SPR: The operator must diagnose the event and then apply information from the stem to arrive at the answer.

Reference:
AOP-2B Unit 1, Feedwater System Malfunction, Rev 19, Step 8
ARP 1C03 1E2 1-2, Steam Generator A (B) Level Setpoint Deviation/Trouble Rev 1, Step 3.2

Proposed reference to be provided to the applicants during examination:
None

Justification:

The annunciators and MFP control switch indications convey that a single main feed pump has tripped. The condition is addressed by AOP-2B, Feedwater System Malfunction which directs a power reduction to less than 60% per AOP-17A, Rapid Power Reduction if operating at less than 75% power.

A **INCORRECT:** AOP-2B directs load reduction to a point that a single MFP can support operation. Action is correct if the initiating power level is at or above 75%.

B **CORRECT:** See above explanation.

C **INCORRECT:** AOP-2B directs load reduction to a point that a single MFP can support operation. This response is plausible if student determines that the loss of feed requires removing secondary load. The turbine can be tripped if below P-9, and actions to stabilize would be per AOP-25, Turbine Trip without reactor trip

D **INCORRECT:** ARP 1CO3 1E2 1-2 (1-5) Steam Generator A (B) Level Setpoint Deviation/Trouble provides directions to restore the controller to AUTOMATIC if the problem was a controller issue.

Learning Objective:

Describe the procedures which govern operation of the Feedwater System. (052.05.LP0128.005)

5. 2017 NRC 080/EPE/055EG2.2.37/4.6/1-B/SRO/BANK/TS B 3.7.6/057.01.LP3336.017

Which of the following components has a Technical Specification function which is necessary to mitigate the effects of a Station Blackout (SBO) from Rated Thermal Power?

A. SI Accumulators have a minimum volume and pressure to make up for RCS leakage and losses until power is restored to ECCS pumps.

B. Pressurizer Power Operated Relief Valves nitrogen backup supply prevents cycling of the Primary Safety Valves during the heat-up phase of a SBO.

C. Diesel Fuel Oil system automatic transfer of fuel to maintain minimum level in the day tanks allowing diesel operation for the duration of the blackout coping time.

D.✓ Condensate Storage Tanks (CSTs) have a minimum inventory to maintain MODE 3 for the blackout coping time before needing to transfer suction to Service Water.

SRO Tier 1 Group 1

Source: Bank

Question History:
None

Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "question refers to plant conditions existing prior to the Unit 2 refueling outage."

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

K/A:
055EG2.2.37 Station Blackout
Ability to determine operability and/or availability of safety related equipment. (Imp 3.6/4.6)

Cognitive Level:
1-B Memory: The student must recall the information from tech spec bases for which functions mitigate a Station Blackout.

Reference:
TS B 3.7.6, Condensate Storage Tanks (CST), Rev 5, Safety Analyses and Bases Sections

Proposed reference to be provided to the applicants during examination:
None

Justification:

A **INCORRECT:** This is the answer for the worst case small break LOCA analyses. Plausible if the student compounds a LOCA with a SBO, since during an SBO there is no source of injection to the RCS.

B **INCORRECT:** The PORVs have nitrogen lined up to them when they are used during LTOP, not during normal operations. Plausible if the student assume the PORVs will function after the loss of air.

C **INCORRECT:** This is the answer for the AC sources-Shutdown. There is a surveillance which demonstrates ability to move fuel oil to the day tanks. Plausible if the student determines that the cure to the SBO is to start a diesel.

D **CORRECT:** The limiting event for CST volume is the SBO event. The minimum amount of water is the CST assures the capability to maintain the unit in MODE 3 for at least an hour concurrent with a loss of all AC power, while then allowing sufficient operator action time for transfer of AFW suction to the service water system.

Learning Objective:

DISCUSS Technical Specification Definitions, Rules of Usage, Safety Limits, 1 Hour or Less Actions for Systems, Equipment, and Bases of LCOs and Safety Limits.
(057.01.LP3336.017)

6. 2017 NRC 081/EPE/E05EA2.2/4.3/3-SPK/SRO/NEW/CSP-H.1/043.03LP1998.007

Given the following:

- Unit 1 was at Rated Thermal Power
- A LOCA occurred causing an automatic Safety Injection
- The crew has transitioned to CSP-H.1, Response to Loss of Secondary Heat Sink, and is checking if a secondary heat sink is required.
- Plant conditions are as follows:
  - RCS Pressure – 1170 psig and STABLE
  - RCS Hot Leg Temperature 535°F and RISING SLOWLY
  - Pressurizer Level – 0%
  - SG ‘A’ Wide Range Level – 40 inches and LOWERING SLOWLY
  - SG ‘B’ Wide Range Level – 105 inches and LOWERING SLOWLY
  - Containment pressure – 6 psig and RISING SLOWLY
  - RWST Level – 58% LOWERING SLOWLY
  - AFW Flow – 0 gpm

What action(s) will the OS1 initially direct at this time?

A✓ Transition to EOP-1.3, Transfer to Containment Sump Recirculation – Low Head Injection.

B. Transition to EOP-1.4, Transfer to Containment Sump Recirculation – High Head Injection.

C. Continue in CSP-H.1 to establish AFW flow until directed to return to procedure and step in effect.

D. Continue in CSP-H.1, establish RCS Bleed and Feed while continuing efforts to restore feed flow.

SRO Tier 1 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

K/A:
E05EA2.2 Loss of Secondary Heat Sink
Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): **Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.**
(Imp 3.7/4.3)

Cognitive Level:
Comprehension 3-SPK: The student must understand the initial conditions, use procedural knowledge to determine if foldout page criteria is met, and rules of usage for determining the correct action.

Reference:
CSP-H.1 Unit 1, Response to Loss of Secondary Heat Sink, Rev 42,
Foldout Page Item 2

Proposed reference to be provided to the applicants during examination:
None

Justification:

The current plant conditions tell the student that, adverse containment conditions exist, there is a small break LOCA occurring, based on RCS pressure, entry into CSP-H.1 is due to low level in the SG and AFW flow less than 230 gpm, a heat sink is required based on SG pressure (determined by RCS hot leg temp and using the steam tables) being less than RCS Pressure and RCS Thot leg temperature being greater than 350°F. Given this procedure is in effect, and the RWST level is less than 60% foldout page item 2 requires transition to EOP-1.3.

A **CORRECT:** See above explanation.

B **INCORRECT:** Transition to EOP-1.3 is required by the foldout page. Plausible if the student determines a transfer to Sump Recirc – High Head is necessary based on RCS pressure being greater than 450 psig and RHR flow being 0 gpm.

C **INCORRECT:** Transition to EOP-1.3 is required by the foldout page. Plausible if the student determines the procedure is in effect and recalls that both steam generator levels need to be less than 85 inches for the given conditions to initiate Bleed and Feed, and does not recall there is a transition for establishing sump recirc prior to establishing a heat sink.

D **INCORRECT:** Transition to EOP-1.3 is required by the foldout page. Plausible if the student does not recall the foldout page requirement based on RWST level and determines that only one SG being less than 85 inches is necessary to require Bleed and Feed to be established.

Learning Objective:

APPRAISE and PRIORITIZE each operator initiated recovery technique in its ability to restore the Heat Sink Critical Safety Function.
(043.03.LP1998.007)

7. 2017 NRC 082/APE/032AA2.07/3.4/2-DR/SRO/NEW/TS 3.3.1/057.01LP3336.011
Given the following:

- Unit 1 is in MODE 3
- Actions are being taken to transition the unit to MODE 2
- Initial conditions of OP 1B, Reactor Startup, were performed during the previous shift
- Due to unforeseen delays, PBF-2034, Control Room Logs Unit 1, is being performed again
- Source Range N-31, N-32 Channel Check failed due to channel deviation
- Both Source Range NI audio circuits are functioning
- The balance of the PBF-2034 has been completed satisfactory

Which of the following states if Unit 1 can enter MODE 2 and the reason?
(Assume a risk assessment will NOT be performed at this time)

A. Yes
Source Range NIs will no longer be applicable in MODE 2 above P-6

B. Yes
Provided that the channel check with either N-31 or N-32 and N-40, WR Source Count Rate is still met

C.✓ No
The required number of Source Range NI channels are not available

D. No
Mode change can not be made until one Source Range NI channel is placed in trip per the SOP

SRO Tier 1 Group 2

Source: New

Question History:
None

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

K/A:
032AA2.07 Loss of Source Range Nuclear Instrumentation
Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Maximum allowable channel disagreement. (Imp 2.8/3.4)

Cognitive Level:
Comprehension 2-DR: The operator must take the information from the stem, and apply it to recalled information multiple times to answer the question.

Reference:
TS 3.3.1, Reactor Protection System (RPS) Instrumentation, Rev 4
TS B 3.3.1, Bases for Reactor Protection System (RPS) Instrumentation, Rev 8, Applicable Safety Analyses, LCO and Applicability Section
0-SOP-IC-002, Technical Specifications LCO-Instrument Cross Reference, Rev 24, Attachment A, Page 15 of 40, Attachment B pages 124 and 125

Proposed reference to be provided to the applicants during examination:
None

Justification:

With the SR NI Channel Check being unsat, one of the two instruments is no longer operable, given that, MODE 2 cannot be entered for the following reasons, The SR NI are applicable in the next mode, there is no allowance for indefinite operation in the next mode, nor is there an allowance or specification stated which will allow operation in MODE 2 with only 1 SR NI

A **INCORRECT:** Entering MODE 2 and the reason is incorrect. Plausible if the student misinterprets the requirements of LCO applicability, as the SR NIs are not required in MODE 2 above P-6, but there is a time when the plant will be in MODE 2 below P-6, therefore the 2 SR NI channels are required.

B **INCORRECT:** Entering MODE 2 and the reason is incorrect. Plausible if the student incorrectly determines that N-40 is an allowable substitute for the SR NIs in MODE 2. When in MODE 6, 2 of the 3 SR NIs can be utilized to meet the LCOs which use SR NIs.

C **CORRECT:** See above explanation.

D **INCORRECT:** The reason is incorrect. Plausible if incorrectly applies the MODE 2 instrument requirements, of placing the affected channel in trip within one hour.

Learning Objective:

Given a copy of Technical Specifications and a set of plant conditions, APPLY the Limiting Condition for Operation (LCO) Applicability requirements of TS Section 3.0.
(057.01.LP3336.011)

8. 2017 NRC 083/APE/060G2.1.28/4.1/1-B/SRO/MODIFIED/EPIP- 1.2.1/SD86.1 2.1.28
Given the following:

- Both units are at Rated Thermal Power
- A Waste Gas decay tank begins to leak
- RE-214, Aux Building Vent Exhaust monitor, goes into HIGH ALARM
- RE-315, Aux Building Exhaust Low-Range monitor, goes into HIGH ALARM
- The Shift Manager declares UNUSUAL EVENT, RU1.2 - VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes or longer

(1) Which of the following correctly describes the FINAL PAB Ventilation Filter alignment assuming no operator actions?
AND
(2) Which of the following indicates the reason an Unusual Event is declared and the Emergency Plan implemented for these conditions?

F-23, PAB Charcoal Filter
F-29, PAB HEPA Filter

A✔ (1) F-23 is Aligned and F-29 is Secured
(2) The release indicates a degradation in plant control and a potential degradation in the level of safety.

B. (1) F-23 is Secured and F-29 is Aligned
(2) The release indicates a degradation in plant control and a potential degradation in the level of safety.

C. (1) F-23 is Aligned and F-29 is Secured
(2) Full Site assembly, accountability, and evacuation of unnecessary personnel is required to protect plant personnel.

D. (1) F-23 is Secured and F-29 is Aligned
(2) Full Site assembly, accountability, and evacuation of unnecessary personnel is required to protect plant personnel.

SRO Tier 1 Group 2

Source: Modified

Question History:

2007 PBNP NRC RO 23

SRO

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:

060G2.1.28 Accidental Gaseous Radwaste Release

Knowledge of the purpose and function of major system components and controls. (Imp 4.1/4.1)

Cognitive Level:

Knowledge 1-F: The operator must understand the initial conditions, recall the automatic actions, and the reason for the EAL classification

Reference:
RMSASRB CI RE-214, Radiation Monitoring System Alarm Setpoint & Response Book Channel Information Sheet, Auxiliary Building Vent Exhaust Gas Monitor, Rev 9
EPIP- 1.2.1, Emergency Action Level Technical Basis, Rev 19, RU1.2

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Consider the following plant conditions:

- *Both units are at 100% power.*
- *PAB Ventilation is aligned as follows:*
  - *W-35, PAB Supply fan, is running.*
  - *W-30A, PAB Filter fan, is running, W-30B is secured.*
  - *W-21A, PAB Stack fan, is running, W-21B is secured.*
  - *PAB Ventilation filters are in the normal alignment.*
- *Subsequently, a Waste Gas decay tank begins to leak.*
- *RE-214, Aux Building Vent Exhaust monitor, goes into HIGH ALARM.*

Which of the following correctly describes the FINAL PAB Ventilation system alignment assuming no operator actions?

	W-21/W-30 PAB Exhaust Fans	*W-35, PAB Supply Fan*	*F-23 PAB Charcoal Filter*	*F-29 PAB HEPA Filter*
A	*As-is*	*Running*	*Aligned*	*Secured*
B	*As-is*	*Running*	*Secured*	*Aligned*
C	*All Running*	*Off*	*Aligned*	*Secured*
D	*All Running*	*Running*	*Aligned*	*Aligned*

Proposed Answer: A

Justification:

F-23, PAB Charcoal filter, will automatically align and F-29, PAB HEPA filter, will automatically secure.

This classification addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for 60 minutes. The fundamental basis of this classification is NOT a dose or dose rate, but rather the degradation in the level of safety of the plant implied by the uncontrolled release

A **CORRECT:** See above explanation.

B **INCORRECT:** Plausible as this is the normal alignment for the ventilation filters, and the basis is correct.

C **INCORRECT:** Plausible as this is the correct alignment for the ventilation filters, and this is the basis for actions which may taken at an ALERT or higher classification

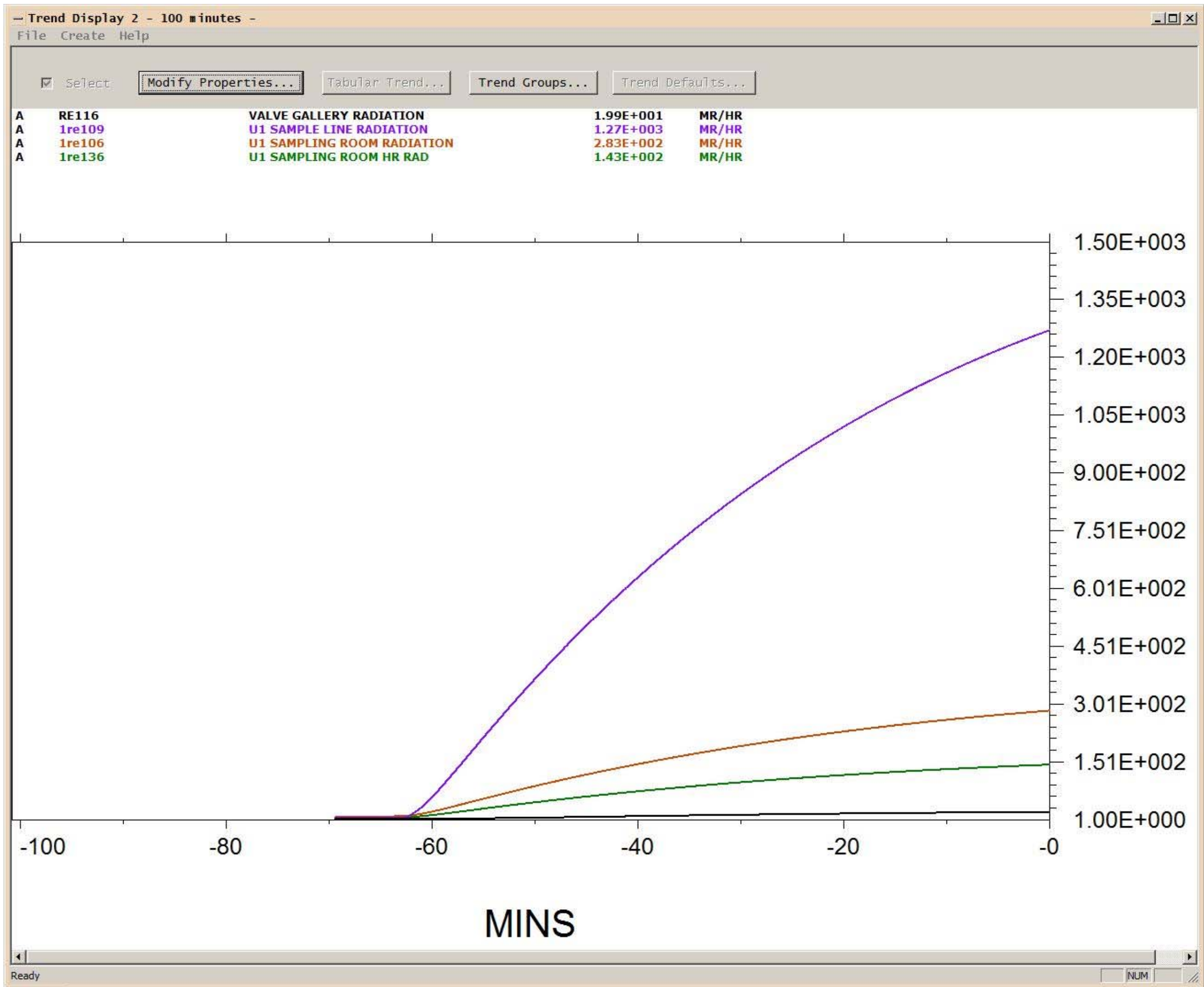
D **INCORRECT:** Plausible as this is the normal alignment for the ventilation filters, and this is the basis for actions which may taken at an ALERT or higher classification.

Learning Objective:

Knowledge of the purpose and function of major system components and controls
(SD86.1 2.1.28)

9. 2017 NRC 084/APE/076AA2.04/3.0/3-SPR/SRO/MODIFIED/AOP-8A/055.03.LP2443.004
Given the following:

- A down power from 100% to 60% was performed in the last 70 minutes due to a Feedwater Control problem
- The following annunciator lit:
    - 1C20 C 1-1, UNIT 1 AREA RADIATION MONITOR HIGH
- Below is a PPCS trend



- Dose-Equivalent Iodine 131 is approximately 100 μCi/gm
- The crew has entered AOP-8A, High Reactor Coolant Activity

Question continued on next page

9. 2017 NRC 084/APE/076AA2.04/3.0/3-SPR/SRO/MODIFIED/AOP-8A/055.03.LP2443.004

Question continued from previous page

Which of the following:
(1) describes the action(s) that will be directed?
AND
(2) identifies the required Technical Specification actions?

A.✓ (1) Raise Letdown flow to 80 GPM;
(2) Plant must be shutdown and cooled down to MODE 3 in 6 hours

B. (1) Raise Letdown flow to 80 GPM;
(2) Plant operation may continue with increased RCS sampling frequency

C. (1) Place 1CV-145, Letdown Divert Temp Control Valve to DIVERT
(2) Plant must be shutdown and cooled down to MODE 3 in 6 hours

D. (1) Place 1CV-145, Letdown Divert Temp Control Valve to DIVERT;
(2) Plant operation may continue with increased RCS sampling frequency

SRO Tier 1 Group 2

Source: Modified

Question History:
2011 Ginna (Retake) NRC SRO 85

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

K/A:
076AA2.04 High Reactor Coolant Activity
Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Process effluent radiation chart recorder. (Imp 2.6/3.0)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, determine which rad monitor is important given the initial conditions, determine the reading of the monitor. Using that gathered data, determine the required actions to direct and the tech spec requirements also.

Reference:
AOP-8A Unit 1, High Reactor Coolant Activity, Rev 12, Step 1, 5, and 11
TS 3.4.16, RCS Specific Activity, Rev 5, Actions C

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given the following:

- *A load reduction from 100% to 60% was performed in the last 30 minutes due to a Feedwater Control problem.*
- *The following alarm is received:*
  - *R-9 LETDOWN MONITOR*
- *Chemistry sample indicates that the high activity is due to failed fuel.*
- *Dose-Equivalent Iodine-131 is approximately 62 microcuries per gram.*
- *Gross degassed activity indicates 87 microcuries per gram.*
- *The crew enters AP-RCS.3, High Reactor Coolant Activity.*

Which ONE of the following describes the action(s) that will be performed in accordance with AP-RCS.3, and identifies the required Technical Specification actions?

A: Raise Letdown flow to 60 GPM;
Plant shutdown and cooldown to less than 500°F must be performed
B: Raise Letdown flow to 60 GPM;
Plant operation may continue with increased RCS sampling frequency
C: Isolate Letdown;
Plant shutdown and cooldown to less than 500°F must be performed
D: Isolate Letdown and place Excess Letdown in service;
Plant operation may continue with increased RCS sampling frequency

Proposed Answer: A

Justification:

Given the initial conditions, the student must determine which rad monitor is the one of note from the trend report, and also based on the trend, what actions are required to be taken. Based on 1RE-109 being greater than 750 mrem/hr and valid, based on both the rise of 1RE-109, but also based on the rise of the other monitors, that both a rise in letdown flow is needed, as well as a shutdown of the plant, based on dose equivalent iodine. Tech specs will require a shutdown of the plant, MODE 3 in 6 hours.

A **CORRECT:** See above explanation.

B **INCORRECT:** The continued operation of the plant is incorrect. Plausible if the student incorrectly recalls the limit for Dose Equivalent 1-131 and confuses it with the Dose Equivalent for I-133 which is 300 μCi/gm.

C **INCORRECT:** The valve operation is incorrect. Plausible if the student attempts to minimize the spread of effects of the failed fuel by diverting the flow of letdown around the demineralizers.

D **INCORRECT:** Valve operation and continued plant operation is incorrect. Plausible if the student incorrectly recalls the limit for Dose Equivalent 1-131 and confuses it with the Dose Equivalent for I-133 which is 300 μCi/gm and attempts to minimize the spread of effects of the failed fuel by diverting the flow of letdown around the demineralizers.

Learning Objective:

Given access to the Site Specific Simulator or specific plant conditions, DEMONSTRATE the ability to respond to the following events in accordance with appropriate procedures:

a. Various Abnormal/High activity levels
b. Material Spills
c. Seismic Event

(055.03.LP2443.004)

10. 2017 NRC 085/EPE/E10G2.4.50/4.0/3-SPK/SRO/NEW/ARB 1C04 1C 1-5/SD86.4 2.4.50

Given the following:

- A substation perturbation caused the loss of 1A01 and 1A02, 4160 VAC Non-Safeguards busses
- The crew is implementing EOP-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS)
- The crew is performing the "Continue RCS Cooldown And Initiate Depressurization" step, and is cooling down to 350°F and depressurizing the plant to 1200 psig
- Power has been restored to 1A01, 4160 VAC Non-Safeguards bus
- The crew starts 1P-1A, Reactor Coolant Pump
- Annunciator 1C04 1C 1-5, 1P-1A&B RCP VIBRATION ALARM is received
- The CO reports 'A' RCP vibrations are as follows:
  - Shaft vibration is 20 mils and SLOWLY RISING
  - Frame vibration is 7 mils and SLOWLY RISING

Which of the following describes the procedure flow path the SRO should direct?

A. Transition to OP 3C, Hot Standby to Cold Shutdown, and commence the plant cooldown to cold shutdown

B✓ Remain in EOP-0.3, and perform AOP-1B, Reactor Coolant Pump Malfunction in parallel and trip the 'A' RCP

C. Remain in EOP-0.3 until the cooldown and depressurization is completed then transition to OP 3C, Hot Standby to Cold Shutdown, and continue the plant cooldown to cold shutdown.

D. Transition to EOP-0.1, Reactor Trip Response, to terminate the cooldown and stabilize the plant with the RCP running, then transition to OP 3D, Post Reactor Trip Stabilization to Hot Standby

SRO Tier 1 Group 2

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
E10G2.4.50 Natural Circulation with Steam Void in Vessel with/without RVLIS
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (Imp 4.2/4.0)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, and then determine what the appropriate procedure flow path he should direct
Reference:
ARB 1C04 1C 1-5, 1P-1A&B RCP VIBRATION ALARM, Rev 4, step 6.3
AOP-1B Unit 1, Reactor Coolant Pump Malfunction, Rev 24, Foldout Item 1
EOP-0.3 Unit 1, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS), Rev 27, Step 1

Proposed reference to be provided to the applicants during examination:
None

Justification:

EOP-0.3, step 1 is a continuous action step to try to restart an RCP, when this is successful, a transition OP 3C is directed for further plant recovery. The annunciator received indicates an off-normal condition with the RCP exists, and directs entry to AOP-1B if necessary which will be performed in parallel while still controlling the cooldown and depressurization. Based on the values for the vibration and given they are also still rising, this meets the foldout page criteria for tripping the RCP. Stating the RCP therefore has not been successful, so transition from EOP-0.3 is not warranted.

A **INCORRECT:** Plausible if the student does not recall the trip criteria of the RCP, and utilizes the normal procedure transition given a successful RCP start.

B **CORRECT:** See above explanation.

C **INCORRECT:** Plausible if the student incorrectly applies the procedure usage to complete the plant cooldown/depressurization based on usage of EOP-0.3 is for a cooldown depressurization to cold shutdown, with no accident in progress.

D **INCORRECT:** Plausible if the student in under the assumption that with restart of an RCP, that the cooldown is no longer required, with no accident in progress, and wants to transition to the post trip procedures.

Learning Objective:

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual
(SD86.4 2.4.50)

11. 2017 NRC 086/SYS/004A2.28/4.3/3-SPK/SRO/NEW/EOP-3/031.02.LP0473.008

Given the following:

- While preparing to shut down, a Steam Generator Tube Rupture occurred on the ‘B’ Steam Generator, along with a loss of 1X-02, Unit 1 Auxiliary Transformer
- The crew entered EOP-3, Steam Generator Tube Rupture
- The crew was preparing to depressurize the RCS to minimize break flow and refill the Pressurizer, but due to equipment malfunctions, PORVs or Auxiliary Spray were **NOT** able to be utilized
- Plant conditions are as follows:
  - Containment pressure is 0.2 psig and STABLE
  - RCS Subcooling is 70°F and STABLE
  - ‘B’ Steam Generator Narrow Range Level is pegged HIGH at 100%
  - ‘B’ Steam Generator Wide Range Level is 405 inches and FLAT LINED
  - ‘B’ Steam Generator Pressure is 1010 psig and RISING SLOWLY
  - The TSC states equipment repair will take an extended amount of time

Based on these conditions, which of the following states:
(1) the procedure transition that will be directed by the OS
AND
(2) the method used to depressurize the RCS?

A. (1) ECA-3.1, SGTR with Loss of Reactor Coolant-Subcooled Recovery Desired
(2) Dump Steam from the ruptured Steam Generator

B. (1) ECA-3.1, SGTR with Loss of Reactor Coolant-Subcooled Recovery Desired
(2) Lower charging flow and establish letdown flow to initiate backflow

C. (1) ECA-3.3, SGTR without Pressurizer Pressure Control
(2) Dump Steam from the ruptured Steam Generator

D.✓ (1) ECA-3.3, SGTR without Pressurizer Pressure Control
(2) Lower charging flow and establish letdown flow to initiate backflow

SRO Tier 2 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

K/A:
004A2.28 Chemical and Volume Control
Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Depressurizing of RCS while it is hot. (Imp 3.7/4.3)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, realize the impact of the inability to depressurize and based on that event, choose the correct procedures to mitigate the consequences and describe the method which will be used in that procedure.

Reference:
EOP-3 Unit 1, Steam Generator Tube Rupture, Rev 51, Steps 17, 18, and 19 RNO
ECA-3.3 Unit 1, SGTR without Pressurizer Pressure Control, Rev 30, Note for Step 31 and Step 31

Proposed reference to be provided to the applicants during examination:
None

Justification:

The initial conditions put the student in EOP-3 as the depressurization step, given no “normal” method of depressurization available, (i.e., spray, PORVs or aux spray) a transition to ECA-3.3 is warranted. After transition the method of depressurization will be to lower charging, raise letdown, while maintaining subcooling.

A **INCORRECT:** The procedure transition is incorrect, and the method for depressurization is also incorrect for the given situation. Plausible, as this transition is in numerous places in EOP-3, and also on the foldout page. The method for depressurization is also contained in ECA-3.1.

B **INCORRECT:** The procedure transition is incorrect. Plausible, as this transition is in numerous places in EOP-3, and also on the foldout page.

C **INCORRECT:** The method for depressurization is incorrect for the given situation. Plausible because this method for depressurization is found in both ECA-3.1 and ECA-3.3, but given the “full” steam generator it is not allowed.

D **CORRECT:** See above explanation.

Learning Objective:

Given access to the Site Specific Simulator and appropriate plant/system conditions, IMPLEMENT the ECAs in response to each of the following events:

a. SGTR in faulted Steam Generator
b. SGTR with a LOCA
c. SGTR with a loss of Pressurizer Pressure control

(031.02.LP0473.008)

12. 2017 NRC 087/SYS/010A2.02/3.9/3-SPK/SRO/MODIFIED/EOP-0/051.01.LP0457.004

Given the following:

- Unit 1 is at 20% of Rated Thermal Power
- The controlling Pressurizer Pressure Channel failed HIGH
- The crew is implementing AOP-24, Response to Instrument Malfunction
- All operator actions to shut 1RC-431A, PZR Spray Valve A Loop were unsuccessful, and it remains stuck partially OPEN
- Pressurizer pressure is 2205 psig and LOWERING

Based on the indications, OS1 will direct the crew to…

A✓ manually trip the reactor, implement EOP-0, Reactor Trip or SI, transition to EOP-0.1, Reactor Trip Response and stop RCP ‘A’.

B. manually trip the reactor, implement EOP-0, Reactor Trip or SI, transition to EOP-0.1, Reactor Trip Response and stop both RCPs.

C. trip RCP ‘A’ per AOP-24, Response to Instrument Malfunction, verify all PZR heaters energized and shut down the reactor within 6 hours.

D. trip RCP ‘A’ per AOP-1D, Chemical and Volume Control Malfunction, verify all PZR heaters energized and shutdown the reactor within 6 hours.

SRO Tier 2 Group 1

Source: Modified

Question History:
2011 Braidwood NRC SRO 87

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

K/A:
010A2.02 Pressurizer Pressure Control System (PZR PCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Spray valve failures. (Imp 3.9/3.9)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial conditions, realize determine what the impact of the fault is going to be, and then choose the correct procedures to mitigate the consequences

Reference:
EOP-0 Unit 1, Reactor Trip or Safety Injection, Rev 64, Step 4 RNO
EOP-0.1, Reactor Trip Response, Rev 45, Step 5 RNO

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Given:
- Unit 1 is at 20% power.
- All systems are normally aligned.
- PZR pressure is 2205 psig and lowering.
- 1RY455C, PZR Spray Valve, is open and NOT responding in manual OR auto.
- All other PZR system components are operating as designed.

Based on the above indications, the Unit Supervisor will direct the crew to…

A. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SI, and STOP the 1C RCP.

B. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SI, and STOP the 1D RCP.

C. STOP the 1C RCP per BwOP RC-2, SHUTDOWN OF A RCP and verify all PZR heaters energized.

D. STOP the 1D RCP per BwOP RC-2, SHUTDOWN OF A RCP and verify all PZR heaters energized.

Answer: A

Justification:

The initial conditions will cause pressure to continually lower until a reactor trip occurs given no operator actions. If actions to stabilize the plant are unsuccessful, as noted in the question stem, the crew should trip the reactor. Transition to EOP-0.1 will be due to no SI having occurred. The guidance for stopping the RCP is in EOP-0.1, Reactor Trip Response, during the RCS pressure trend check.

A **CORRECT:** See above explanation.

B **INCORRECT:** Transition to EOP-0.1 is correct, but stopping both RCPs is not. Plausible if the student misapplies the procedure for securing the RCPs and recall the step having closed bullets, where all actions listed should be taken.

C **INCORRECT:** Entry conditions are correct for AOP-24, but this procedure will not dictate the mitigation of the event. Plausible as this is a procedure the crew will enter, and steps included in the procedure are to establish manual control as required and return affected parameters to desire vales. The 6 hour shutdown is TS required for TS 3.4.4 RCS Loops – MODES 1 and 2.

D **INCORRECT:** Entry conditions are met with a pressurizer pressure low annunciator Plausible as there are steps in the AOP to mitigate this event, i.e., secure a RCP, but those steps are for a RCP cooling issue, not a failed spray valve. The 6 hour shutdown is TS required for TS 3.4.4 RCS Loops – MODES 1 and 2.

Learning Objective:

DESCRIBE the procedures which govern operation of the Pressurizer Pressure and Level Control System. Description should include significant prerequisites, precautions, and notes associated with each operating procedure requiring consideration by Licensed and Non Licensed Operators.
(051.01.LP0457.004)

13. 2017 NRC 088/SYS/039A2.04/3.7/3-SPK/SRO/BANK/CSP-S.1/043.03.LP1996.012

Given the following:

- A TAVE channel failed HIGH causing rod motion
- Taking the Rod Bank Selector switch to MANUAL did **NOT** stop rod motion
- The reactor trip was **NOT** successful
- The turbine trip **WAS** successful
- Reactor trip and bypass breakers are **NOT** able to be opened
- Reactor Power is 65% and LOWERING SLOWLY

Which of the following states the status of the Condenser Steam Dumps and the procedure selected by the OS to mitigate the event?

A. The Condenser Steam Dumps are OPEN and EOP-2, 'Faulted Steam Generator Isolation' will shut the MSIV's to address the faulted S/G concern.

B. The Condenser Steam Dumps are SHUT and EOP-0.1, 'Reactor Trip Response' will control S/G pressure utilizing the Condenser Steam Dumps in pressure control mode.

C✓ The Condenser Steam Dumps are OPEN and CSP-S.1, 'Response to Nuclear Power Generation/ATWS' will shut the MSIV's minimizing the cooldown to address reactivity effects.

D. The Condenser Steam Dumps are SHUT and EOP-0, 'Reactor Trip or Safety Injection' will direct use of the Condenser Steam Dumps to control RCS temperature in the desired band after transition out of CSP-S.1, 'Response to Nuclear Power Generation/ATWS'.

SRO Tier 2 Group 1

Source: Bank

Question History:
2011 PBNP NRC SRO 88

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

K/A:
039A2.04 Main and Reheat Steam System (MRSS)
Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Malfunctioning steam dump.** (Imp 3.4/3.7)

Cognitive Level:
Comprehension 3-SPK: The Student must understand the initial conditions, realize the impact of the failure of the reactor trip, and what transition/procedure will mitigate the event.

Reference:
CSP-S.1 Unit 1, Response to Nuclear Power Generator/AWTS Rev 40, Step 9.
883D195 Sh 17, Steam Dump Control Logic Diagram, Rev 20

Proposed reference to be provided to the applicants during examination:
None

Justification:

CSP-S.1 will be the correct procedure transition based on the unsuccessful reactor trip, EOP-0, immediate action 1 RNO will direct this transition. Condenser steam dumps will be open based on the failure and CSP-S.1 will direct the closing of them during the step to Check for Reactivity Insertion from Uncontrolled RCS Cooldown

A **INCORRECT:** The condenser steam dumps are open due to the failure. Plausible if the student misdiagnoses a faulted S/G due to uncontrolled dumping of steam

B **INCORRECT:** The condenser steam dumps are open due to the fault, EOP-0.1 control of RCS temperature will control the dumps. Plausible if the student misdiagnoses the procedure transition due to no perceived fault with the dumps.

C **CORRECT:** See above explanation.

D **INCORRECT:** The condenser steam dumps are open due to the failure. This is the correct use of the steam dumps for temperature control. Plausible if the student bases answer on proper use of steam dumps and not diagnosis of the plant.

Learning Objective:

LIST the major actions accomplished by the Subcriticality Critical Safety Function Procedures.
(043.03.LP1996.012)

14. 2017 NRC 089/SYS/062G2.2.44/4.4/3-SPK/SRO/BANK/AOP-18A/055.03.LP2440.002

Given the following:

- Unit 2 is in MODE 3
- The following annunciators have been received:
  - C02F 1-4, UNIT 2 4.16 KV BUS UNDER VOLTAGE
  - C02F 1-5, UNIT 2 4.16 KV BUS MAIN OR TIE BREAKER TRIP
  - C02F 2-7, UNIT 2 480V BUS UNDER VOLTAGE
  - C02F 3-4, 4.16 KV BUS LOCKOUT
- G-03 and G-04 are both running unloaded.

Complete the following statement:
___(1)___ is completed as the top priority, and ___(2)___ is the next priority.

AOP-18, Electrical System Malfunction
AOP-18A Unit 2, Train 'A' Equipment Operation
AOP-19A Unit 2, Train 'A' Safeguards Bus Restoration
AOP-19B Unit 2, Train 'B' Safeguards Bus Restoration

A. (1) AOP-18
(2) AOP-19A Unit 2

B.✔ (1) AOP-18A Unit 2
(2) AOP-19B Unit 2

C. (1) AOP-19B Unit 2
(2) AOP-18

D. (1) AOP-19B Unit 2
(2) AOP-18A Unit 2

SRO Tier 2 Group 1

Source: Bank

Question History:
2007 PBNP NRC SRO 90

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

K/A:
062G2.2.44 A.C. Electrical Distribution
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. (Imp 4.2/4.4)

Cognitive Level:
Comprehension 3-SPK: The Student must understand the initial conditions, interpret the annunciators, and then determine which procedures will mitigate the event, and also which priority to perform the procedures in.

Reference:
ARB C02 F 1-4, UNIT 2 4.16 KV BUS UNDER VOLTAGE, Rev 7, Step 6.5
ARB C02 F 1-5, UNIT 2 4.16 KV BUS MAIN OR TIE BREAKER TRIP, Rev 7, Step 6.8.3
ARB C02 F 2-7, UNIT 2 480V BUS UNDER VOLTAGE, Rev 5, Step 6.2
ARB C02 F 3-4, 4.16 KV BUS LOCKOUT, Rev 6,
AOP-18A Unit 2, Train ‘A’ Equipment Operation, Rev 17, Step 27
AOP-19B Unit 2, Train ‘B’ Safeguards Bus Restoration, Rev 12, Step 1

Proposed reference to be provided to the applicants during examination:
None

Justification:

With the given conditions, 2A-06, 4160kV Safeguards Bus is locked out. AOP-18A is the first priority, and AOP-19B is the next. If AOP-19B is entered first, the first step has the operator carry out the action of AOP-18A to align train ‘A; equipment.

A **INCORRECT:** AOP-18 addresses a non-vital low voltage condition or X04 lockout, so AOP-18 is not the proper procedure to enter. Plausible if the student misdiagnoses the event based on given annunciators.

B **CORRECT:** See above explanation.

C **INCORRECT:** AOP-19B will be used to restore power to the ‘B’ train buses, but ‘A’ train alignment is checked prior to doing this. Plausible if the student prioritizes restoration over alignment. AOP-18 addresses a non-vital low voltage condition or X04 lockout, so AOP-18 is not the proper procedure to enter. Plausible if the student misdiagnoses the event based on given annunciators.

D **INCORRECT:** Priorities are reversed; AOP19B step 1 has the operator carry out the actions AOP-18A prior to taking any further actions in AOP-19B. Plausible if the student reverses procedure priority.

Learning Objective:

Given access to the Site Specific Simulator or specific plant conditions, RESPOND to the following conditions:

a. Turbine Generator Voltage Regulator failure
b. Loss of Main Generator Hydrogen pressure
c. Total collapse of 345 KV system frequency
d. Loss of electrical buses

(055.03.LP2440.002)

15. 2017 NRC 090/SYS/076G2.4.21/4.6/3-SPK/SRO/NEW/CSP-ST.0/043.03.LP1995.013
Given the following:

- A reactor trip and Safety Injection occurred as a result of a large break LOCA about 90 minutes ago
- EOP-1.3, Transfer to Containment Sump Recirculation – Low Head Injection is in progress
- Containment Spray has been aligned for recirculation and the crew is aligning charging pump suction to the VCT
- The following plant conditions are noted:

• Reactor Coolant System Pressure	15 psig	STABLE
• Core Exit Thermocouples	186°F	LOWERING SLOWLY
• RCS Subcooling	60°F	RISING SLOWLY
• Containment pressure	8 psig	LOWERING SLOWLY
• Steam Generator ‘A’ Level	55%	STABLE
• Steam Generator ‘B’ Level	58%	STABLE
• Containment Sump 'B' level	90 inches	RISING SLOWLY
• RWST Level	17%	STABLE
• Containment Radiation Monitor	31 R/hr	RISING SLOWLY
• Reactor Vessel Narrow Range Level	17 ft	STABLE
• Reactor Vessel Wide Range Level	45 ft	STABLE
• Service Water Header –South	35 psig	STABLE
• Service Water Header – North	55 psig	STABLE
• Reactor Coolant Pumps are secured		

Which of the following actions should be taken in response to these indications?

A. Stay in EOP-1.3, Transfer to Containment Sump Recirculation – Low Head Injection, CSP’s may not be implemented while this procedure is in progress.

B. Enter ECA-1.3, Containment Sump Blockage, based on high Containment Sump B level.

C. Enter CSP-P.1, Response to Imminent Pressurized Thermal Shock, and take actions including a 1 hour RCS temperature soak.

D.✓ Enter CSP-Z.2, Response to Containment Flooding, to address containment flooding and direct U2 OS to implement AOP-9A, Service Water Malfunction in parallel.

SRO Tier 2 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
076G2.4.21 Service Water System (SWS)
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (Imp 4.0/4.6)

Cognitive Level:
Comprehension 3-SPK: The operator must understand the initial condition, which one of the conditions must be calculated by the operator, and apply those conditions to the critical safety function status Trees, and interpret how the information impacts the EOP Network and determine the course of action to take next.

Reference:
CSP-ST.0 Unit 1, Critical Safety Function Status Trees, Rev 9, Step 4, Figure 5

Proposed reference to be provided to the applicants during examination:
None

Justification:

Given the initial conditions, the crew has aligned containment spray for recirculation, the student must determine where in EOP-1.3 the crew currently is, and then based on that, determine if a transition out of EOP-1.3 to a CSP is allowed. This is based on the note at Step 1 “Steps 1 through 32 should be performed without delay. CSPs should not be implemented prior to completion of these steps.” The crew is currently on step 36. The AOP-9A actions are necessary to mitigate the high sump level to protect critical plant components necessary for plant recovery. Service Water indication provides further basis for addressing containment flooding.

A **INCORRECT:** The step in effect in EOP-1.3 is past the note which allows implementation of CSPs. Plausible because EOP-1.3 suspends implementation of CSPs for most of the procedure.

B **INCORRECT:** ECA-1.3 is incorrect because the indication provided do not result from sump blockage, and lowering CETs provide positive indication that sump recirculation is effective. This response is plausible if the examinee has a misconception that sump blockage could result in rising containment level.

C **INCORRECT:** Plausible as the temperature soak is normally the method utilized to mitigate the excessive cooldown. This will not be performed because RCS pressure is 15 psig, and since the crew is in EOP-1.3, RHR flow is greater than 550 gpm, so CSP-P.1 will be exited at step 1, and the crew will return to the procedure in effect; not taking the soak actions.

D **CORRECT:** See above explanation.

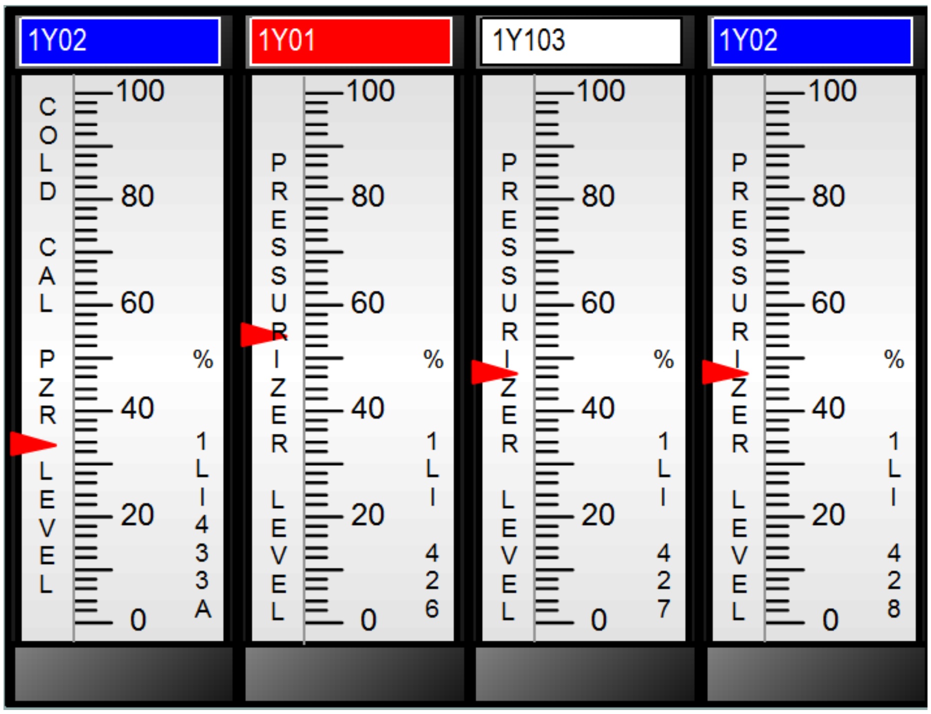
Learning Objective:

IMPLEMENT the Critical Safety Function Status Tree and Critical Safety Procedure rules of usage.
(043.03.LP1995.013)

16. 2017 NRC 091/SYS/016G2.2.42/4.6/2-DR/SRO/NEW/TS B 3.3.1/057.02.LP3341.002

Given the following:

- Unit 1 is at Rated Thermal Power
- Pressurizer Level indicates as shown



Complete the following statements:
The REQUIRED ACTIONS of placing the channel in trip per Tech Spec 3.3.1, Reactor Protection System (RPS) Instrumentation, Function 8, Pressurizer Water Level – High, ___(1)___ required to be performed at this time for long-term continued operation at power.

And

The Tech Spec 3.3.1 Bases for the High Pressurizer Water Level reactor trip is to provide ___(2)___.

Question continued on next page

16. 2017 NRC 091/SYS/016G2.2.42/4.6/2-DR/SRO/NEW/TS B 3.3.1/057.02.LP3341.002

Question continued from previous page

A. (1) ARE
(2) a backup for OPΔT

B.✔ (1) ARE
(2) a backup for the Pressurizer Pressure-High trip

C. (1) are NOT
(2) a backup for OPΔT

D. (1) are NOT
(2) a backup for the Pressurizer Pressure-High trip

SRO Tier 2 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

K/A:
016G2.2.42 Non-Nuclear Instrumentation System (NNIS)
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (Imp 3.9/4.6)

Cognitive Level:
Comprehension 2-DR: The student must interpret the meter reading, and then recall information requirement multiple mental steps.

Reference:
PBF-2034, Control Room Log – Unit 1, Rev 93, Page 111
TS B 3.3.1, Bases for Reactor Protection System (RPS) Instrumentation, Rev 8, Applicable Safety Analyses Section 8 page 16

Proposed reference to be provided to the applicants during examination:
None

Justification:

With the channel difference being greater than the allowable of 4%, the channel check would be considered unsat, and that channel would be OOS. Therefore the actions of T.S. 3.3.1 would be taken. Per the basis document the Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip.

A **INCORRECT:** The signal does not provide a backup for OP∆T. Plausible if the student has the misconception that as pressure rises (over power) temperature rises, causing PZR level to rise, as well as pressure due to the insurge, and the fact that OP∆T provides a signal to generator a turbine runback which can in turn cause a high pressurizer pressure/water level.

B **CORRECT:** See above explanation

C **INCORRECT:** Tech Specs actions should be performed as the channel is outside of the allowable limit and the signal does not provide a backup for OP∆T. Plausible if the student misinterprets the allowable channel difference and has the misconception that as pressure rises (over power) temperature rises, causing PZR level to rise, as well as pressure due to the insurge, and the fact that OP∆T provides a signal to generator a turbine runback which can in turn cause a high pressurizer pressure/water level

D **INCORRECT:** Tech Specs actions should be performed as the channel is outside of the allowable limit. Plausible if the student misinterprets the allowable channel difference.

Learning Objective:

Given specific plant conditions, ASSESS and APPLY Technical Specification requirements as appropriate.
(057.02.LP3341.002)

17. 2017 NRC 092/SYS/035G2.4.41/4.6/3-SPR/SRO/NEW/EPIP-1.2/SD86.4 2.04.41

Given the following:

- Unit 1 is at Rated Thermal Power
- An RCS Leak occurs
- The crew commences a 3%/min down power
- With Pressurizer Level at 38% and LOWERING SLOWLY it is decided to manually trip the reactor due to not being able to maintain Pressurizer level
- The crew transitions to EOP-0, Reactor Trip or Safety Injection
- The Reactor trip pushbuttons on 1C04 fail to work
- The Reactor trip pushbuttons on C01 successfully trip the reactor
- The crew has transitioned from EOP-0, Reactor Trip or Safety Injection to the appropriate procedure, plant conditions are as follows:
    - Core Exit Thermocouples are 650°F LOWERING SLOWLY
    - Containment Pressure 6 psig is RISING SLOWLY
    - Both Reactor Coolant Pumps have been tripped
    - Reactor Vessel Narrow Range is 12 ft
    - SG ‘A’ AF Flow 0 gpm
    - SG ‘B’ AF Flow 0 gpm
    - SG ‘A’ Narrow Range is 0%
    - SG ‘B’ Narrow Range is 5%

Using the provided reference, which of the following describes the HIGHEST emergency classification and action level for this event?
(See provided references)

A. Alert based on SA2

B✓ Site Area Emergency based on FS1

C. Site Area Emergency based on SS4

D. General Emergency based on SG2

Note - the following additional information was provided to applicants during the exam in response to a question that was asked: "add the following bullet to the information in the stem of the question: RCS pressure is 1400 psig and lowering slowly."

SRO Tier 2 Group 1

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
035G2.4.41 Steam Generator (S/GS)
Knowledge of the emergency action level thresholds and classifications.
(Imp 2.9/4.6)

Cognitive Level:
Comprehension 3-SPR: The operator must understand the initial conditions and sequence of events, and apply those and the current indication to provided reference to determine what the highest classification is.

Reference:
EPIP 1.2 , Emergency Classification, Rev 53, Fission Product Barriers
EPIP 1.2.1, Emergency Action level Technical Basis, Rev 19, Basis Information for Table F-1 Page 85
CSP-ST.0 Unit 1, Critical Safety Function Status Trees, Rev 9, Figure 3

Proposed reference to be provided to the applicants during examination:
EPIP 1.2 , Emergency Classification, Page 9 and 11
EPIP 1.2.1, Emergency Action Level Technical Basis, Pages 70, 77, and 81

Justification:

Based on the initial conditions, with the first transition to CSP-H.1, the Critical Safety Function Status Trees need to be monitored. With entry conditions to CSP-H.1 met, the classification will be FS1 based on a potential loss of fuel clad barrier and potential loss of RCS barrier.

A **INCORRECT:** Not the highest classification. Plausible based on this being a correct answer, if the first trip button failure is viewed as an auto trip not occurring, but this is not the highest classification for the situation.

B **CORRECT:** See above explanation

C **INCORRECT:** Incorrect action level. The classification (SAE) is correct, the EAL would be correct if CSP-C.1 and CSP-H.1 entry conditions are met. Plausible if the student incorrectly recalls the basis for SS4, and confuses it with CSP-C.2, which is met.

D **INCORRECT:** Incorrect action level. The action level would be correct if the reactor was not tripped from the control room due to also meeting the entry conditions for CSP-H.1. Plausible if the student has a misconception of what constitutes the basis of SG2 (the ATWS requirement).

Learning Objective:

Knowledge of the emergency action level thresholds and classifications. (2.04.41)

18. 2017 NRC 093/SYS/072A2.03/2.9/3-SPR/SRO/NEW/RECM/053.05.LP0286.006
Given the following:

- Unit 1 is at Rated Thermal Power
- Unit 2 is in MODE 5 cooling down for a refueling outage
- A discharge per OI-140C, Standard Radioactive Batch Liquid Release – Monitor Tanks is in progress
- Fuse failure in C-257, RMS DAM-07 Control Panel causes a loss of power to:
  - RE-101, Control Room Monitor
  - RE-214, Auxiliary Building Vent Exhaust Gas Monitor
  - RE-218, Waste Disposal System Liquid Monitor
  - RE-220, Spent Fuel Pool Cooling Water Liquid Monitor
  - RE-225, Combined Air Ejector Low Range Monitor

PAB exhaust automatically shifts filters, and . . .
(See provided references)

3.3.5 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions inoperable.	A.1 Place CREFS in the emergency mode of operation.	7 days

A. - CREFS will remain in MODE 1.
- WL-18 Waste Cond Ovbd Disch to SW Hdr remains open.
- LCO 3.3.5 is MET.
- The liquid discharge may continue based on RE-229 Service Water Discharge being operable.

B. - CREFS will remain in MODE 1.
- WL-18 Waste Cond Ovbd Disch to SW Hdr remains open.
- Enter LCO 3.3.5 ACTION CONDITION A.
- The liquid discharge must be stopped manually and may not be restarted per ODCM.

C. - CREFS will shift to MODE 5.
- WL-18 Waste Cond Ovbd Disch to SW Hdr trips shut.
- LCO 3.3.5 is MET due to being in the failure mode.
- The liquid discharge may be restarted based on RE-229 Service Water Discharge being operable.

D.✓ - CREFS will shift to MODE 5.
- WL-18 Waste Cond Ovbd Disch to SW Hdr trips shut.
- Enter LCO 3.3.5 ACTION CONDITION A.
- The liquid discharge may not be restarted per ODCM.

18. 2017 NRC 093/SYS/072A2.03/2.9/3-SPR/SRO/NEW/RECM/053.05.LP0286.006

SRO Tier 2 Group 2

Source: New

Question History:
None

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

K/A:
072A2.03 Area Radiation Monitoring (ARM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **Blown power-supply fuses.** (Imp 2.7/2.9)

Cognitive Level:
Comprehension 3-SPR: The operator must understand the initial conditions, determine what effect the loss of power has had on the plant, and what TS and ODMC issues apply, determine the correct actions necessary.

Reference:
ODCM, Offsite Dose Calculation Manual, Rev 19, Table 6-2
TS 3.3.5, Control Room Emergency Filtration System (CREFS) Actuation Instrumentation Rev 2
1-SOP-Y-Y104, 1Y-104, Yellow 120V Vital Instrument Panel, Rev 6, Section 5.3.6, and Attachment I
PBF-2068G – DAM-7, Status of an RMS Channel Not in Service, Rev 5, Discussion Section

Proposed reference to be provided to the applicants during examination:
ODCM Tables 6-2 AND 7-2 (pages 31 and 43)

Justification:

Given the power loss, the following will happen; PAB ventilation will shift to through the charcoal filters, the liquid discharge will stop, and CREFS will shift to mode 5 (RO knowledge). The student must further determine the TS implications, which based on the loss of the rad monitor, condition A must be entered, and using the ODCM, the discharge is not allowed to be restarted with the loss of RE-218 (SRO knowledge).

A **INCORRECT:** The auto actions are incorrect and action are incorrect. Plausible if the student incorrectly recalls a loss of power to a rad monitor will cause the automatic actions NOT to happen, based on those actions not happening the TS is plausible, as it is still in the normal mode with an another rad monitor still in service, and the liquid discharge passed RE-229 and did not lose power.

B **INCORRECT:** The auto actions are incorrect but the TS and ODCM actions are correct. Not the highest classification. Plausible if the student incorrectly recalls a loss of power to a rad monitor will cause the automatic actions NOT to happen, based on the loss of the rad monitor TS entry is correct, the action to manually stop the liquid discharge is plausible, because the ODCM does not allow the discharge to continue.

C **INCORRECT:** The auto actions are correct but the TS and ODCM actions are incorrect. Plausible if the student correctly recalls automatic actions, and based on those action determines the TS is met due to being in the failed condition and the liquid discharge possible due to going passed RE-229 which did not lose power.

D **CORRECT:** See above explanation

Learning Objective:

ASSESS the effects of malfunctions in the Radiation Monitoring System.
(053.05.LP0286.004)
IDENTIFY and DISCUSS the Technical Specifications associated with the Radiation Monitoring System components, parameters, and operation to include:
a. Limiting Condition for Operation (LCO)
b. LCO Applicability
(053.05.LP0286.006)

19. 2017 NRC 094/GEN/2.1.2/4.4/1-F/SRO/BANK/OM 3.26/SD86.1 2.1.2
Given the following:

- Both Units are in MODE 1
- Inservice testing activities will require the following:
  - A Dedicated Operator in the PAB to maintain equipment availability
  - A Level 2 Assigned Operator in the Control Room to operate a Unit 1 containment isolation valve

Assuming all operators are fully qualified, which of the following can the SRO direct per OM 3.26, Use of Dedicated / Assigned Operators?

A.✓ Assign the CO3 Watchstander to be the Level 2 Assigned Operator.

B. Assign the CO1 Watchstander to be the Level 2 Assigned Operator.

C. Assign the Water Treatment AO to be the Dedicated Operator in the PAB.

D. Assign a relief crew AO, who is a member of the fire brigade, to be the Dedicated Operator in the PAB.

SRO Tier 3

Source: Bank

Question History:
None

SRO
The performance of assignment/direction of Dedicated/Assigned Operators is an SRO task based on OM 3.26, The shift manager is responsible for:
1) Ensuring the use of a dedicated / assigned operator is appropriate for the Situation,
2) Determining the level of dedicated / assigned operator required and the qualifications necessary to perform each specific operator assignment.

K/A:
2.1.2 Conduct of Operations
Knowledge of operator responsibilities during all modes of plant operation.
(Imp 4.1/4.4)

Cognitive Level:
Knowledge 1-F: The operator must recall the requirements of in the use of dedicated and assigned operators.

Reference:
OM 3.26, Use of Dedicated / Assigned Operators, Rev 16, Sect 4.2.9
OM 3.41, System Status Control, Rev 9, Attachemt B, 4.1.9

Proposed reference to be provided to the applicants during examination:
None

Justification:
Per 4.2.9, Watchstanders other than the "Operator at the Controls" may be designated as a Level 2 Assigned Operator provided the operator is directly performing the evolution in progress and remains at the valve while it is open. The CO1 Watchstander is the Operator at the Controls for Unit 1.

A **CORRECT:** See above explanation.

B **INCORRECT:** Per 4.2.9: Watchstanders other than the "Operator at the Controls" may be designated as a Level 2 Assigned Operator provided the operator is directly performing the evolution in progress and remains at the valve while it is open.

C **INCORRECT:** Per 4.4.5, For actions performed outside the control room the dedicated operator must remained stationed at the proper location for these actions during the test and NOT have a designated watch station OR be part of the fire brigade complement.

D **INCORRECT:** Per 4.4.5, For actions performed outside the control room the dedicated operator must remain stationed at the proper location for these actions during the test and NOT have a designated watch station OR be part of the fire brigade complement.

Learning Objective:
Knowledge of operator responsibilities during all modes of plant operation. (SD86.1 2.1.2)

20. 2017 NRC 095/GEN/2.1.6/4.8/1-P/SRO/MODIFIED/OM 3.7/031.03.LP0158.004
Given the following:

- The crew is implementing the EOP network
- Plant conditions are requiring prompt actions to mitigate damage to a component in the form of a variance (non-deviation)

At a MINIMUM, which of the following requirements must be met?

A. - Action should be approved by two cognizant on-shift SROs if time permits
- Notification to NRC required prior to performance of actions
- Action, time and reason for the variance logged

B. - Action must be approved by one cognizant on-shift SRO and Operations Director
- Action, time and reason for the variance logged
- One-hour notification made to the NRC

C. - Action must be approved by Assistant Operations Manager - Shift
- Action cannot violate the intent of the procedure
- Action, time and reason for the variance logged

D.✓ - Action should be approved by two cognizant on-shift SROs if time permits
- The variance is not a departure from any license condition or technical specification
- Action, time and reason for the variance logged

SRO Tier 3

Source: Modified

Question History:
None

SRO
10CFR55.43(b)(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

K/A:
2.1.6 Conduct of Operations
Ability to manage the control room crew during plant transients.
(Imp 3.8/4.8)

Cognitive Level:
Knowledge 1-P: The operator must recall the requirements of procedure when invoking a variance.

Reference:
OM 3.7, AOP and EOP Procedure Usage for Response to Plant Transients, Rev 28, Sections 6.9.3, and 6.21

Proposed reference to be provided to the applicants during examination:
None

Original Question:
Due to plant conditions requiring prompt actions to mitigate damage to the RCPs, an Emergent Change is being considered to the EOP in use.

At a MINIMUM, which of the following requirements must be met?
(DCS is Duty and Call Supervisor; CAP AR is a Corrective Action Program Action Request.)

A. - Action must be approved by one cognizant on-shift SRO and the DCS notified.
- Action, time and reason for the Emergent Change logged.
- One-hour notification made to the NRC.

B. - Action must be approved by two cognizant on-shift SROs and the DCS.
- Action cannot violate the intent of the procedure.
- Action, time and reason for the Emergent Change logged.
- CAP AR and Procedure Change Request initiated as soon as plant conditions allow.
- One-hour notification made to the NRC.

C. - Action must be directed by an SRO and approved by the Shift Technical Advisor.
- CAP AR and Procedure Change Request initiated as soon as plant conditions allow.
- Action, time and reason for the Emergent Change logged.

D. - Action must be approved by two cognizant on-shift SROs.
- Action cannot violate the intent of the procedure.
- CAP AR and Procedure Change Request initiated as soon as plant conditions allow.
- Action, time and reason for the Emergent Change logged.

Proposed Answer: D

Justification:

Per section 6.9 and 6.21, the following is the minimum required:
- Action should be approved by two cognizant on-shift SROs if time permits
- Prior to authorizing a procedure variance, the SRO in command shall determine whether the variance is also a deviation.
- If the variance was a deviation, the requirements of 10 CFR 80.72(b.1) SHALL be observed
- The Procedure variance that was taken SHALL be logged.
- Action taken using the procedure variance or variance deviation process and station logged entries SHALL be documented using an Action Request after the actions are taken.

A **INCORRECT:** See above explanation. Plausible as notification of the NRC prior to action is a desired thing.

B **INCORRECT:** See above explanation. Plausible based on the approval by a senior licensee management person is a desired thing and notification being required.

C **INCORRECT:** See above explanation. Plausible as these are all things that are required for changes under different circumstances.

D **CORRECT:** See above explanation.

Learning Objective:

DESCRIBE when it is appropriate to deviate from EOP guidance and what actions are necessary.
(031.01.LP0158.004)

21. 2017 NRC 096/GEN/2.2.7/3.6/1-P/SRO/BANK/EN-AA-203-1202/SD86.2 2.2.7
When a test or experiment is proposed which may affect the PBNP License or Technical Specifications, the activity is scrutinized using a multi-phase process.

Which part of this process DETERMINES whether PBNP must obtain NRC approval PRIOR to carrying out the test or experiment?

A. 10CFR50.59 Screening

B✓ 10CFR50.59 Evaluation

C. 10CFR50.59 Amendment

D. Onsite Review Group (ORG) Review

SRO Tier 3

Source: Bank

Question History:
2012 PBNP NRC Exam SRO 95

SRO
10CFR55.43(b)(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

K/A:
2.2.7 Equipment Control
Knowledge of the process for conducting special or infrequent tests.
(Imp 2.9/3.6)

Cognitive Level:
Knowledge 1-P: The operator must recall the requirements of the procedure when invoking a test or experiment.

Reference:
EN-AA-203-1202, 10 CFR 50.59 Evaluation, Rev 1, Section 4.1.1.B

Proposed reference to be provided to the applicants during examination:
None

Justification:

A 10 CFR 50.59 Evaluation is written to compare the activity against the 10 CFR 50.59(c)(2) criteria to determine whether prior NRC approval is required.

A **INCORRECT:** The screening concludes that a portion or portions of the proposed activity “screen in” for further evaluation under 10 CFR 50.59. Plausible as this portion of the process looks at if this activity is going to need further attention from the process.

B **CORRECT:** See above explanation.

C **INCORRECT:** A license amendment may be required based on the outcome of this process, but the amendment is not part of the 10CFR50.59 evaluation. Plausible, as this is a logical outcome of the process.

D **INCORRECT:** The Onsite Review Group is part of the 10CFR50.59 process and reviews the evaluation paperwork and UFSAR changes. Plausible as this group is utilized after the determination has been made that a changes to the UFSAR is needed, this would be an activity requiring prior approval.

Learning Objective:

Knowledge of the process for conducting special or infrequent tests.
(SD86.2 2.2.7)

22. 2017 NRC 097/GEN/2.2.21/4.1/1-P/SRO/BANK/MA-AA-203-1000/SD86.2 2.2.21
Given the following:

- Unit 1 is at Rated Thermal Power
- The packing was replaced on 1SI-860B, 1P-14A CS Pump Reduced Flow Discharge MOV
- 1SI-860B was left in the CLOSED position
- The Clearance Order Removal is complete and power has been restored to the MOV

Which of the following states the minimum action(s) required to restore 1SI-860B to OPERABLE after the valve packing replacement?

1SI-860B may be restored to OPERABLE status when . . .

(RMP, Routine Maintenance Procedure)

A. the MOV's auto open function is tested per its Surveillance Test Procedure.

B.✓ the MOV has been satisfactorily time stroked per its Surveillance Test Procedure.

C. the MOV is stroked open and closed from the control room per the guidance of the applicable RMP.

D. the MOV is manually stroked open and closed per the applicable RMP guidance, with no packing leakage verified.

SRO Tier 3

Source: Bank

Question History:
2013 Farley NRC Exam SRO 94

SRO
10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

K/A:
2.2.21 Equipment Control
Knowledge of pre- and post-maintenance operability requirements.
(Imp 2.9/4.1)

Cognitive Level:
Knowledge 1-P: The operator must understand the initial conditions, the extent of the necessary repairs and then recall what needs to be done to make the valve operable again.

Reference:
MA-AA-203-1000, Maintenance Testing, Rev 7, Attachment 3, Mechanical Components Test Matrices, Page 30 of 32 and Attachment 6, Mechanical Testing Guide and Test Definitions, Page 64 of 69
RMP 9376-1, Limitorque MOV Removal/Installation/Swap and Testing or Gate and Globe Valves, Rev 19, Step 5.11.13 and 5.13.8
IT 05 Train A, Train A Containment Spray Pump and Valves Unit 1, Rev 2, Section 1.2 and Attachment B

Proposed reference to be provided to the applicants during examination:
None

Justification:

Based on the repair, and the fact that the breaker or motor were not worked on, the only retests to make the valve operable again, are those which ensure the valve is capable of meeting its required function. No test of an auto feature is needed, but a stroke time is.

A **INCORRECT:** The auto open function is not required to be tested. Plausible as this test is required if the electrical portion of the valve was affected by the repair.

B **CORRECT:** See above explanation.

C **INCORRECT:** This is done, but does not make the valve operable. Plausible, as this manipulation of the valve is done, and it will prove that the valve has freedom of movement.

D **INCORRECT:** This is done, but does not make the valve operable. Plausible as this manipulation of the valve is done, and it will prove that the valve has freedom of movement.

Learning Objective:

Knowledge of pre- and post-maintenance operability requirements.
(SD86.2 2.2.21)

23. 2017 NRC 098/GEN/2.3.13/3.8/1-P/SRO/BANK/RP-AA-103.1002/SD86.3 2.3.13
Given the following:

- The following radiological conditions exist for an area in the plant:
    - General dose rates range from 25-45 mrem/hr
    - Measurements taken on pipes and valves are:
        - Point 1 is 100 mrem/hr at 30 cm
        - Point 2 is 500 mrem/hr at 30 cm
        - Point 3 is 1100 mrem/hr at 30 cm
- A Non-Licensed Operator (NLO) needs to enter this area to isolate a safety related system during Emergency Operating Procedure implementation
- No RP Supervisor can be reached

Based on the above information which of the following states:
(1) the radiological posting required for this area
AND
(2) who can authorize entry by the NLO?

A. (1) Very High Radiation Area
(2) Only a RP Supervisor

B. (1) Very High Radiation Area
(2) The Plant General Manager and RP Manager

C. (1) Tech Spec Locked High Radiation Area
(2) Only the Plant General Manager

D.✓ (1) Tech Spec Locked High Radiation Area
(2) The Shift Manager if no RP Supervisor can be reached

SRO Tier 3

Source: Bank

Question History:

2012 Beaver Valley Unit 1 NRC 98

SRO

10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

K/A:

G2.3.13 Radiation Control

Knowledge of radiological safety 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, aligning filters, etc. (Imp 3.4/3.8)

Cognitive Level:

Knowledge 1-P: The operator must recall the requirements for posting and entry in to that area.

Reference:

RP-AA-103-1002, High Radiation Area Controls, Rev 6, Sections 2.17, and 4.3.1

Proposed reference to be provided to the applicants during examination:

none

Justification:

Per RP-AA-103-1002, a Tech Spec Locked High Radiation Area (LHRA) is defined as RCA areas where general area radiation level exceeds 1000 millirem per hour (Deep Dose Equivalent) at 30 cm (11.8 in) from a radiation source or surface that radiation penetrates.

Per RP-AA-103-1002, 4.3 RP Requirements to Allow Access to Tech Spec Locked High Radiation Areas

1. If urgent need arises to access a Locked High Radiation Area and RP Supervision is not available, THEN PEFRORM the following:
   A. OBTAIN approval from Shift Manager to enter LHRA
   B. NOTIFY RP Supervisor as soon as practical

A **INCORRECT:** Both halves are incorrect. Plausible if the student confuses 500 R/hr with millirem/hr requirements, and has the misconception that only an RP Supervisor can grant access which is a normal method during non-urgent times.

B **INCORRECT:** Both halves are incorrect. Plausible if the student confuses 500 R/hr with millirem/hr requirements, and this is the correct list of people who can grant access to a Very High Radiation Area.

C **INCORRECT:** The second half is incorrect. Plausible if the student has the misconception of the PGM or RP Supervisor granting access and the RP Supervisor is not available.

D **CORRECT:** See above explanation.

Learning Objective:

Knowledge of radiological safety 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, aligning filters, etc
(SD86.3 2.3.13)

24. 2017 NRC 099/GEN/2.4.25/3.7/1-P/SRO/MODIFIED/NP 1.9.14/SD86.4.2.4.25

A fire is reported to the Control Room by an Office Assistant and verified by an Auxiliary Operator in the area of the Lube Oil Storage Room.

According to NP 1.9.14, 'Fire Protection Organization,' which of the following is **NOT** a responsibility of the Shift Manager regarding fire emergency response guidelines?

The Shift Manager should perform all of the following except . . .

A. implement the Emergency Plan.

B. contact the duty and call superintendent (DCS) as soon as possible.

C.✓ proceed to the scene of the fire to direct activities, after assuring another SRO is in the control room.

D. contact the Two Creeks Volunteer Fire Department for assistance as soon as fire magnitude is known.

SRO Tier 3

Source: Modified

Question History:
2011 PBNP NRC SRO 99

SRO
10CFR55.43(b)(1) Conditions and limitations in the facility license.

K/A:
2.4.25 Emergency Procedures / Plan
Knowledge of fire protection procedures. (Imp 3.3/3.7)

Cognitive Level:
Knowledge 1-P: The operator must recall the requirements of the procedure in question.

Reference:
NP 1.9.14, Fire Protection Plan, Rev. 19, Section 4.10.5.c and d

Proposed reference to be provided to the applicants during examination:
None

Original Question:
A fire is reported to the Control Room by an Office Assistant and verified by an Auxiliary Operator in the area of the Lube Oil Storage Room.

According to NP 1.9.14, 'Fire Protection Organization,' which of the following describes a responsibility of an Operating Supervisor regarding fire emergency response guidelines?

The responsible Operating Supervisor should . . .

A. proceed to the scene to act as the fire brigade leader, after assuring the Shift Manager is in the control room.

B. implement the Emergency Plan..

C. contact the Two Creeks Volunteer Fire Department for assistance as soon as fire magnitude is known.

D. relieve the Shift Manager who will proceed to the scene of the fire to direct activities.

Proposed answer: A

Justification:

NP 1.9.14 Fire Protection Plan delineates the duties of the control room watchstanders and other site personnel in responding to a fire on-site.

A **INCORRECT:** The shift manager is to proceed to the control room and take actions required to assure proper event reporting. He is the ERO Emergency Director until relieved by TSC or EOF staff.

B **INCORRECT:** Per NP 1.9.14, the Shift Manager is to notify the duty and call superintendent as soon as feasible

C **CORRECT:** If designated as the the Fire Brigade Leader, the Operating Supervisor would assure the SM has been notified to come to the control room and would proceed to the fire scene after ensuring the SM or other SRO is in the control room.

D **INCORRECT:** Per NP 1.9.14, the Shift Manager is to contact the Two Creeks Volunteer Fire Department for assistance, based on available information or Fire Brigade Leader request.

Learning Objective:

Knowledge of fire protection procedures
(SD86.4 2.4.25)

25. 2017 NRC 100/GEN/2.4.32/4.0/3-SPR/SRO/NEW/EPIP-1.2/SD86.4 2.4.32

Given the following:

- Both units are at Rated Thermal Power
- A DUAL Unit trip occurs when D01, 125 V DC Bus is de-energized due to a fault
- 60 seconds later the following Unit 1 annunciators are lit:
    - 1C20 C 4-9, 1C20 A & C ANNUNCIATOR POWER FAILURE CKT BRKR 380
    - 1C04 1A 4-1, 1C04 ANNUNCIATOR POWER FAILURE CKT BRKR 52
    - 1C03 1F 4-11, 1C03 ANNUNCIATOR POWER FAILURE CKT BRKR 69
    - C01 C 4-11, C01 ANNUNCIATOR POWER FAILURE CKT BRKR 120
    - C02 D 1-5, G-01 EMERGENCY DIESEL DC POWER FAILURE
    - C02 D 4-5, UNIT 1 SAFEGUARDS DC CONTROL POWER FAILURE
    - C02 D 4-6, UNIT 1 COMMON CRITICAL CONTROL POWER FAILURE
    - C02 E 4-1, UNIT 1 OR 2 UNIT LOCKOUT TRIP CIRCUIT DISABLED
    - C02 E 4-2, UNIT 1 OR 2 TG-01 OR X-01 RELAY TROUBLE
    - C02 F 1-2, G-03 EMERGENCY DIESEL DC POWER FAILURE
- AO reports there is visible damage to D72-1-1, D01 supply breaker

Using the provided reference, which of the following describes the HIGHEST emergency classification for this event?
(See provided references)

A. No classification

B. Unusual Event based on SU3

C.✓ Alert based on SA4

D. Site Area Emergency based on SS6

SRO Tier 3

Source: New

Question History:
None

SRO
10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

K/A:
2.4.32 Emergency Procedures/Plan
Knowledge of operator response to loss of all annunciators. (Imp 3.6/4.0)

Cognitive Level:
Comprehension 3-SPR: The operator must understand the initial conditions, and utilizing this information, determine the correct classification for the event based on the given indications.

Reference:
EPIP 1.2 , Emergency Classification, Rev 53, Hot Initiating Condtions
EPIP 1.2.1, Emergency Action Level Technical Basis, Rev 19, SA4.1

Proposed reference to be provided to the applicants during examination:
EPIP 1.2 , Emergency Classification, Page 9 and 11
EPIP 1.2.1, Emergency Action Level Technical Basis, pages 65, 71-72, and 78

Justification:

Based on the initial conditions, there is a loss of greater than 75% of the annunciators for Unit 1, including the annunciators on C01. Both units have tripped based on the initial conditions, so the classification will be Alert based on SA4, Unplanned loss of most or all safety system annunciator or indication in control room with either (1) SIGNIFICANT TRANSIENT in progress or (2) compensatory non alarming indications are unavailable.

A **INCORRECT:** Plausible based on only 50% of the total control room annunciators are lost.

B **INCORRECT:** Plausible based on the loss of annunciators for Unit 1, but the operator must also recall that the plant will trip, and that trip is defined as a SIGNIFICANT TRANSIEMT.

C **CORRECT:** See above explanation

D **INCORRECT:** Plausible based on the operator assuming the trip is a SIGNIFICANT TRANSIENT, and given the loss of the annunciators, and also the loss of D01, the operator may incorrectly recall the red instruments will be without power, therefore hampering the ability to monitor the transient.

Learning Objective:

Knowledge of operator response to loss of all annunciators
(SD86.4 2.4.32)