U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information

Name:	
Date:	Facility/Unit:
Region: I ☐ II ☐ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	___________	Points
Applicant's Score	___________	Points
Applicant's Grade	___________	Percent

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001.AK3.04
	Importance Rating	3.4

295001 Partial or Complete Loss of Forced Core Flow Circulation- Knowledge of the reasons for the following responses as they apply to partial or complete loss of forced core flow circulation:

AK3.04 Reactor SCRAM

Question: 1

With the Reactor initially operating at 72% power the following conditions exist:

- Reactor Recirculation MG A trips on motor generator high air temperature.
- PMIS shows the plant is operating in the Stability Exclusion Region (red region) of the Power to Flow map.
- APRMs recorders indicate 25% peak to peak power cycling.

Why is a scram required?

A. The large oscillation threshold has been reached.

B. RPV level control is unpredictable due to power swings.

C. It is the fastest way to exit the area of operation while in single loop.

D. The reactor must be shut down prior to exceeding the RPV pressure limit.

Answer:
A. The large oscillation threshold has been reached.

Explanation:

FROM Technical Specification Basis 3.4.1

The reactor is designed such that thermal hydraulic oscillations are prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal hydraulic instability, a Stability Exclusion Region, to be avoided during normal power operation, is calculated using approved methodology. Specific directions are provided to avoid operation in this region and to immediately exit upon entry. Entries into the Stability Exclusion Region are not part of normal operation. An entry may occur as the result of an abnormal event, such as a single recirculation pump trip. In these events, operation in the Stability Exclusion Region may be needed to prevent equipment damage, but actual time spent inside the Region is minimized. Although operator action can prevent the occurrence of and protect the reactor from an instability event, the APRM Neutron Flux High (Flow Biased) scram function will suppress oscillations prior to exceeding the Safety Limit MCPR. While core-wide reactor instability is the predominate mode and regional mode oscillations are not expected to occur, the reactor is protected from regional mode oscillations through avoidance of the Stability Exclusion Region and administrative controls on reactor conditions which are primary factors affecting reactor stability.

Analytical results indicate that the fuel clad may experience boiling transition during this process but that it subsequently rewets and is adequately cooled even for oscillations that resemble reactivity excursion events. For an occasional large pulse, however, rewetting of some of the highest-powered locations within the highest-powered fuel bundles may not occur; the clad could then continue to heat up over several oscillation cycles. The possibility of localized fuel clad failures cannot be precluded even though core geometry and core cooling are not significantly threatened. Propagation of the fuel clad failures to neighboring bundles is not expected and greater core damage is not likely for this condition.

The Large Oscillation Threshold (LOT) is 25% and is a peak-to-peak neutron flux oscillation amplitude sufficiently large to be distinguishable from the flux perturbations expected of a stable thermal-hydraulic system.

From 2.4RR Reactor Recirculation Abnormal:

1. IMMEDIATE OPERATOR ACTIONS

   1.1 IF both RR pumps are tripped and reactor power > 1% rated thermal, THEN perform following.

      1.1.1 **SCRAM**.

      1.1.2 Enter Procedure 2.1.5.

   1.2 IF abnormal neutron flux oscillations are observed while operating in the Stability Exclusion Region, THEN perform following:©[2]

      1.2.1 **SCRAM**.

      1.2.2 Enter Procedure 2.1.5.

   1.3 IF recirculation flow is not stable, THEN perform following:

      1.3.1 IF recirculation flow is rising, THEN perform following:

         1.3.1.1 Press SCOOP TUBE LOCKOUT button.

         1.3.1.2 IF flow still has not stabilized, THEN trip affected RR pump and enter Attachment.

      1.3.2 IF recirculation flow is lowering, THEN press SCOOP TUBE LOCKOUT button.

ATTACHMENT 3 OPERATION IN THE STABILITY EXCLUSION REGION IF operation in Stability Exclusion Region of Power-To-Flow Map, THEN perform following:

1. IF operation in Stability Exclusion Region of Power-To-Flow Map, THEN perform following:

   1.1 Inform Shift Manager LCO 3.4.1, Condition A, entry required.

<u>NOTE</u> 1 – When operating at high rod line and a single recirculation pump, rod insertion may be preferred method to exit Stability Exclusion Region. Core flow response may be small when raising recirculation pump speed.

<u>NOTE</u> 2 – Abnormal neutron flux oscillations are indicated by any of the following:©[2]

- LPRM upscale or downscale indications are alarming and clearing (annunciators or full core display indicators) with an annunciation period of < 3 seconds.
- Sustained rising oscillations on APRMs reaching two or more times its initial peak-to-peak level at reduced core flow (< 50% WT).
- SRM period positive to negative SRM period swings with a characteristic fluctuation time of < 3 seconds.

Distracters:

B. This option is incorrect because there is no requirement to scram based upon RPV level swings. The reactor feedwater pumps have anticipatory circuitry to be able to maintain RPV level in a suitable band so the operators will not think level control is unreliable. This option is plausible because abnormal condition procedure 2.4RXLVL directs scramming the reactor if RPV level cannot be maintained between 12 inches and 50 inches. The candidate who believes the power cycling would cause unreliable RPV level would select this option.

C. This option is incorrect because the direction to scram is not based on the fastest way of exiting the stability exclusion area. The scram is required because the core is exhibiting thermal hydraulic instabilities. Technical Specifications states to immediately exit the region but does not specifically state to scram the reactor. Procedure 2.4RR requires the operator to insert a manual reactor scram based on observed abnormal neutron flux oscillations. This option is plausible because scramming is an immediate way of exiting the exclusion region.

D. This option is incorrect because RPV pressure limits are not threatened with 25% power swings. The RPV high pressure scram is designed to shut down the reactor before the thermal power transferred to the reactor coolant increases and challenges the integrity of the fuel cladding and the reactor coolant pressure boundary. Reactor power protection from the APRM flow biased high flux will occur before the pressure spike reaches the RPV pressure limit scram. This option is plausible because power swings will cause pressure swings.

Technical Reference(s): 2.4RR Reactor Recirculation Abnormal, Rev. 40

Proposed references to be provided to applicants during examination: NONE

Learning Objective: COR002-22-02, OPS Reactor Recirculation System, Rev. 32

5. Briefly describe the following concepts as they apply to the Reactor Recirculation system, or to the Recirculation Flow Control system:

   i. Power to Flow Map (including normal operation/startup/shutdown and Stability Exclusion Region)
   l. Thermal limits

Question Source:	Bank #	______
	Modified Bank #	______
	New	X
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 2

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295003 AA1.03
Importance Rating	4.3

295003 Partial or Complete Loss of AC:

Ability to operate and/or monitor the following as they apply to partial or complete loss of A.C. power:

AA1.03 Systems necessary to assure safe plant shutdown

Question: 2

With the plant operating at 100% power, a loss of offsite power occurs.

- Both diesel generators fail to start and CANNOT be started.
- The Supplemental Diesel Generator is NOT available.
- HPCI and RCIC recover RPV water level to +35 inches (Narrow Range).

What operational restriction applies to the continued use of HPCI in response to this event, until onsite or offsite electrical power can be restored?

HPCI must be secured...

A. just prior to the division's battery being exhausted.

B. after one cycle of operation, and must remain off.

C. NO LATER THAN 15 minutes, from the time that injection flow was reduced, and must remain off.

D. after one cycle of operation, but can be restarted if RPV level lowers to +3 inches narrow range.

Answer:

B. after one cycle of operation and must remain off.

Explanation:

From 5.3SBO

NOTE – Following step ensures compliance with commitment to secure HPCI after one cycle of operation.

1.2 Within ~ 10 minutes of HPCI operation, perform following:Ⓟ[1]

1.2.1 Start RCIC per Procedure 2.2.67.1.

1.2.2 Secure HPCI per Procedure 2.2.33.1.

1.2.3 Ensure HPCI AUXILIARY OIL PUMP to PULL-TO-LOCK.

This procedure assumes that RPV water level and pressure is initially controlled by High Pressure Coolant Injection (HPCI), as directed by the EOPs. CNS has committed to secure HPCI after one cycle of operation, even if EOPs allow HPCI use, in order to extend station battery life during station blackout (SBO). One cycle of HPCI is ~ 10 minutes. SBO analysis assumes Reactor Core Isolation Cooling (RCIC) is operable and maintains RPV level and pressure until on-site or off-site electrical power can be restored. If RCIC is unable to perform this function, compensatory actions must be taken to ensure adequate core cooling. This could include starting HPCI.

Distracters:

A. This option is incorrect because HPCI is manually secured after approximately 10 minutes of operation. Operation until the battery is exhausted would be inconsistent with the commitment to secure HPCI after 1 Cycle of operation. The operator who does not correctly recall the restriction in 5.3SBO to extend station battery life would select this option. This option is plausible because operating HPCI can be performed without the use of a lot of battery power as HPCI turbine speed is generally not changed too much to control injection.

C. This option is incorrect because 15 minutes from the time HPCI flow is reduced is inconsistent with the commitment to operate HPCI for no more than one cycle (~10 minutes) during a Station Black Out event. The operator who does not correctly recall the restriction in 5.3SBO to extend station battery life would select this option. This option is plausible because the utilization of HPCI and/or RCIC during this event may be required, but specific electrical system basis recollection is required.

D. This option is incorrect because HPCI is secured after one cycle of operation and no procedural step allows restarting per 5.3SBO. If the operator does not remember the restriction in 5.3SBO they would select this option. This option is plausible because EOPs allow the use of HPCI to maintain adequate core cooling at a much lower RPV level. There is no reason to be concerned with RPV level at this point because it is at +35 inches (normal band) and RPV level would have to drop ~200 inches to reach the point where adequate core cooling is challenged. In the event that adequate core cooling is challenged, the EOPs override the emergency procedure 5.3SBO.

Technical Reference(s): 5.3SBO Station Blackout, Rev 33.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: INT032-01-31 CNS Abnormal Procedures (RO) Electrical

W. Given plant condition(s) and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #	13338
	Modified Bank #	
	New	
Question History:	Last NRC Exam	2002
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 4

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004.AA2.01
	Importance Rating	3.2

295004 Partial or Total Loss of DC Pwr- Ability to determine and/or interpret the following as they apply to partial or complete loss of D.C. power:

AA2.01 Cause of partial or complete loss of D.C. power

Question: 3

The plant is operating at power with the following conditions:

- Breaker 1FE red indicating light is illuminated.
- Breaker 1GE red indicating light is illuminated.
- 4160V buses A, C, and E indicating lights are off.

What is causing the above conditions?

A. Panel BB1 has a blown fuse.

B. Panel BB3 has a blown fuse.

C. Panel AA1 has a blown fuse.

D. Panel AA3 has a blown fuse.

Answer:

C. Panel AA1 has a blown fuse.

Explanation

Panel AA1 provides DC power to the 4160V buses A, C, and E indication. Breaker 1FE indication is supplied by Panel AA3 and breaker 1GE indication is supplied by Panel BB3.

Distracters:

A. This option is incorrect because Panel BB1 is not the power supply to any of the breakers listed in the question. This option is plausible because Panel BB1 does provide power supply to other 4160V breaker indications in the control room. The candidate who does not correctly recall the power supplies listed would select this option.

B. This option is incorrect because breaker 1GE has indication lights which are powered from BB3. This option is plausible because BB3 does supply other breaker indication in the control room. The candidate who does not correctly recall the power supplies listed would select this option.

D. This option is incorrect because breaker 1FE has indication lights which are powered from AA3. This option is plausible because AA3 does supply other breaker indication in the control room. The candidate who does recognize the divisional power of the breakers but does not correctly recall the power supplies listed would select this option.

Technical Reference(s): APP 2.3_9-5-2 Rev. 43

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR0020702 OPS DC ELECTRICAL DISTRIBUTION

8. Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following:

   p. AC Electrical Distribution

Question Source: Bank # _______
Modified Bank # _______
New ___X____

Question History: Last NRC Exam _______

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 4

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295005.2.1.23
Importance Rating	4.3

295005 Main Turbine Generator Trip

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question: 4

Reactor power is 35% during a startup.

Main Turbine bearing vibrations are as follows:

- Bearing 5 vibration rises rapidly to 15 mils and steadies out.
- Vibration on bearings 4 and 6 are 7 mils and rising.

What action(s) is/are required?

A. Lower reactor power until bearing vibration lowers to <14 mils.

B. Suspend the startup to allow raising the turbine casing temperature.

C. Trip the Main Turbine and enter Procedure 2.2.77 Turbine Generator only.

D. Scram the Reactor and enter Procedure 2.1.5 Reactor Scram, then trip the Main Turbine.

Answer:
D. Scram the Reactor and enter Procedure 2.1.5 Reactor Scram, then trip the Main Turbine.

Explanation:

There are three bearings (4, 5, and 6) that have shown a rise in vibration, and one of them (5) rises above the action point for tripping the Main Turbine. There is no indication given that the number 9 bearing is rising. With reactor power >29.5% (164.5 psig first stage pressure) Annunciator 9-5-2/C-4 is clear. This requires the operator to scram the reactor.

Since there is an unexpected turbine or generator vibration rise, there is an entry condition for procedure 2.4TURB. 2.4TURB Attachment 1 High Vibration is applicable.

NOTE **–** 7 mils causes the indicator bar on TGI-M-DUA and TGI-M-DUB to turn yellow, 10 mils turns TGI-M-DUA and TGI-M-DUB red.

2. Validate vibration by observing vibration on several bearings as read on TGI-M-DUA, TGI VIB/INST MONITOR CHANNEL A, and/or TGI-M-DUB, TGI VIB/INST MONITOR CHANNEL B.

   2.1 Select Turbine Mimic and/or Generator Mimic to determine which bearing is alarming.

   2.2 Select alarmed bearing screen, as required.

   2.3 IF time permits, THEN locally observe turbine for vibration (i.e., visual indication or feeling of abnormal vibration of turbine casing, bearing casing, or attached piping; abnormal sounds; or if local instrumentation is installed, indication of abnormal vibrations indicated).

3. **For Bearings 1 through 8, if rotor vibration ≥ 14 mils coincident with changing vibration levels on at least one other bearing:**

   3.1 **IF Annunciator 9-5-2/C-4 is clear, THEN SCRAM and enter Procedure 2.1.5.**

   3.2 Trip Main Turbine.

   3.3 IF reactor was not scrammed, THEN enter Procedure 2.2.77.

4. For **Bearing 9 only**, IF rotor vibration ≥ **14 mils**, THEN reduce power to maintain < 14 mils.

5. IF any bearing vibration is ≥ 10 mils, THEN immediately contact Turbine Engineering Group for data analysis and recommendation, and Vendor support.

6. IF vibration is ≥ 10 mils on any bearing while operating in DEH MODE 2 (turbine not tied to grid), THEN trip Main Turbine.

7. Review TG parameters to determine if vibration is related to another problem.

8. IF vibration is > 7 mils and < 10 mils, THEN contact Turbine Engineering Group for data analysis and action plan.

Distracters:

A. This option is incorrect because lowering reactor power is only taken if bearing 9 vibration is rising above 14 mils. Because this is the action to take for bearing 9, this option is plausible. The candidate who recalls lowering power to bring bearing vibration down but does not recall it being only for bearing 9 would select this option.

B. This option is incorrect because a reactor scram is required. This is an action to be taken if the rotor becomes long during startup which makes this answer plausible. There is no indication turbine expansion is excessive. The candidate who recalls rotor long actions and believes it is causing high vibrations would select this option.

C. This option is incorrect because a reactor scram is required before tripping the turbine. With the reactor power level given, the reactor is scrammed and then the turbine is tripped. Tripping the turbine and not scramming the reactor would force a reactor scram and operators should not force an automatic RPS trip. The candidate who does not realize the reactor power level is high enough that a turbine trip would cause a reactor scram would select this option.

Technical Reference(s): 2.4TURB Main Turbine Abnormal, Rev. 30
2.3_9-5-2 (Panel 9-5-2 Annunciator Response Procedure), Rev. 43

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective:
INT0320127, CNS Abnormal Procedures (RO) Turbine/Generator

O. Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

P. Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

Question Source:	Bank #	_______	
	Modified Bank #	_24663_	(See attached)
	New	_______	
Question History:	Last NRC Exam	__ _____	
Question Cognitive Level:	Memory or Fundamental Knowledge	_____	
	Comprehension or Analysis	__X__	
10 CFR Part 55 Content:	55.41 (10)		

Comments:

LOD: 3

ORIGINAL QUESTION: 24663

Reactor power is at 17% during a startup. After the turbine is rolling the following occur:

- ☐ Vibration rises abnormally on the Bentley Nevada Display Monitors for bearings 4, 5 and 6.
- ☐ Bearing 5 vibration rises rapidly to 15 mils, then returns to 13 within 30 seconds.

What action(s) is/are required?

a. Trip the Main Turbine and enter Procedure 2.2.77.

b. Lower power per Procedure 2.1.10 while observing vibration.

c. Contact Turbine Engineering Group for data analysis and action plan.

d. Scram the Reactor and enter Procedure 2.1.5, then trip the Main Turbine.

ANSWER: 24663
a. Trip the Main Turbine and enter Procedure 2.2.77.

Explanation:

Since there is an unexpected turbine or generator vibration rise, there is an entry condition for procedure 2.4TURB. Attachment 1 High Vibration step 2 states that if rotor vibration is above 14 mils on one bearing coincident with changing vibration levels on at least one other bearing and Annunciator 9-5-2/C-4 is clear SCRAM and enter Procedure 2.1.5 then trip the Main Turbine and if the reactor was not scrammed, enter Procedure 2.2.77. In this case, there are three bearings that have shown an increase, and one of them (5) rises above the action point for tripping the Main Turbine. With first stage pressure less than 164.5 psig, annunciator 9-5-2/C-4 is clear. So the required action is to perform step 2.2 of attachment 1 and trip the main turbine and then 2.3 enter Procedure 2.2.77.

Distractors:

b. This is not correct because this is the action for vibrations lower than the trip action point. A candidate might choose this answer if they believe that with vibration on bearing 5 lowering to 13 in 30 seconds, might allow not tripping the turbine.

c. Again, this is incorrect because it is an action to be taken before the trip action point is reached.

d. In this case there is no requirement to scram the reactor because annunciator 9-5-2/C-4 is clear because first stage pressure is less than the setpoint.

Examination Outline Cross-Reference: Level RO
Tier # 1
Group # 1
K/A # 295006.AK1.01
Importance Rating 3.7

295006 SCRAM- Knowledge of the operational implications of the following concepts as they apply to SCRAM:

AK1.01 Decay heat generation and removal

Question: 5

The plant is operating at 100% power on the 221st day of continuous operation when all outboard MSIVs go closed.

- The reactor automatically scrams.
- Reactor pressure spikes to 1100 psig.

What is the status of the SRVs 10 minutes after the scram?

A. 1 SRV is cycling.

B. No SRVs are open or cycling.

C. 1 SRV is open and another SRV is cycling.

D. 2 SRVs are open and another SRV is cycling.

Answer:
A. 1 SRV is cycling.

Explanation:

For the first hour after the Reactor Scram decay generation in the reactor is exponentially decreased from approximately from 7% to 1%. Each of the 8 relief valves are designed to flow 9% (power) when at rated pressure. With the MSIVs shut, the decay heat removal is via the Nuclear Pressure Relief system to the torus. Low-Low Set (LLS) arms under these conditions and RV-71 D controls RPV pressure between 875 and 1010 psig.

Distracters:

B. This answer is incorrect because the decay heat load at 10 minutes following a reactor scram requires at least one SRV (RV-71D) to be periodically cycling to control RPV pressure between 875 psig to 1010 psig. This option is plausible if the candidate does not construct an accurate mental model regarding the status of the MSIVs or if the candidate does not remember the amount of decay heat generated following a reactor scram.

C. This option is incorrect because RV-71D is controlling reactor pressure between 875 psig to 1010 psig. If one LLS set valve is not enough to maintain Reactor Pressure then the second LLS valve opens at 1040 psig also cycle to between 1040 and 875. Since there is no indication given of an ATWS, a single relief valve is capable of maintaining reactor pressure. If the candidate does not know that 1 SRV is capable of maintaining reactor pressure based on decay heat rate of 7% they may believe both LLS SRVs are needed. This option is plausible because another SRV is required to maintain pressure if reactor output is above the capacity of 1 SRV.

D. This option is incorrect because RV-71D is controlling RPV pressure between 875 psig to 1010 psig. If one LLS set valve is not enough to maintain Reactor Pressure in the required band, then the second LLS valve opens at 1040 psig and then a 3rd SRV opens at 1090 psig and cycle to maintain RPV pressure. This option is plausible during an ATWS event or if the candidate does not recall that 1 SRV is capable of maintaining reactor pressure based on the decay heat rate of 7%.

Technical Reference(s): 2.2.1 Nuclear Pressure Relief System, Rev.38

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR0021602 Nuclear Pressure Relief

8. Predict the consequences a malfunction of the following would have on the NPR system:

i. Main Steam system

Question Source: Bank # _18065______
Modified Bank # _______
New _______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 55.41 (8)

Comments:

LOD 2

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295016 AK2.01
Importance Rating	4.4

295016 Control Room Abandonment:

Knowledge of the interrelations between control room abandonment and the following: (CFR: 41.7 / 45.8)

AK2.01 Remote shutdown panel: Plant-Specific

Question: 6

Following a toxic gas event requiring the control room to be abandoned, the following conditions exist:

- The MSIVs are closed.
- BOTH Low-Low Set (LLS) valves are cycling.

The ADS ISOLATION switch in the Alternate Shutdown Room is now placed in ISOLATE.

Which LLS valve is able to be controlled from the ASD room?
Which LLS valve continues to cycle?

	Controlled from ASD Room	LLS Valve that Continues to Cycle
A.	RV-71F	RV-71D
B	RV-71F	RV-71F
C.	RV-71D	RV-71D
D.	RV-71D	RV-71F

Answer:

A. RV-71F RV-71D

Explanation:

There are two LLS valves that automatically control reactor pressure once LLS is activated 71D and 71F. The bottom section of the ADS/REC panel contains two isolation switches. One switch is for REC Pumps C and D, the other switch is for Safety Relief Valves 71E, 71F, and 71G. These isolation switches remove valve position and pump status indication from the Control Room and prevent automatic valve operation from ADS or Low Low Set Logic. LLS valve 71D operation is not effected by the ASD panel switch operation.

Distracters:

B. This option is incorrect because Low-Low set valve controlled from the ASD room is not able to cycle on Low-Low set. The candidate would select this option if he/she did not understand that LLS logic has been removed from 71F.

C. This option is incorrect because Low-Low set valve 71D can continue to cycle. The candidate would choose this if he/she thought LLS logic was removed from both 71D and 71F.

D. This option is incorrect because Low-Low set valve 71D can continue to cycle. The candidate would select this if he/she did not understand that LLS logic has been removed from 71F.

Technical Reference(s): GE Electrical Drawing 753E253 Sheet 2.

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR002-34-02 Ops Alternate Shutdown

2. Describe the interrelationship between ASD and the following:
   a. Nuclear Pressure Relief (NPR) system

Question Source: Bank # ________
Modified Bank # _21372_ (See attached)
New ________

Question History: Last NRC Exam ___2006___

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 3

From CNS 2006 NRC Exam

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	21372	00	06/21/2006	10/07/2006	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	COR0023402, Effect that Remote Shutdown actions have on LLS.

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR0023402001020A	Describe the interrelationship between ASD and the following: Nuclear Pressure Relief (NPR) system

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Related Skills (K/A)	
295016.AK2.01	Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8) Remote shutdown panel: Plant-Specific (4.4*/4.5*)

QUESTION: 6 21372 (1 point(s))

Following a toxic gas event requiring the control room to be abandoned, the following conditions existed:

- The MSIVs are closed
- **BOTH** Low-Low Set valves are cycling

The **ADS ISOLATION** switch in the Alternate Shutdown Room is now placed in **ISOLATE.**(NO other actions have been taken outside the control room)

What is the effect on the Low-Low Set valves?

a. Both Low-Low Set valves **stop** cycling.

b. Both Low-Low Set valves **continue** to cycle.

c. The Low-Low Set valve (71F), which **can** be controlled from the ASD room **continues** to cycle.

d. The Low-Low Set valve (71D) which **cannot** be controlled from the ASD room **continues** to cycle.

ANSWER: 6 21372

d. The Low-Low Set valve (71D) which **cannot** be controlled from the ASD room **continues** to cycle.

Reference: COR0023402 ASD Room

EXPLANATION: Placing the Isolation switch in ISOLATE will prevent operation of the Low-Low set valve operated from the ASD room but does not affect the other Low-Low set valve.

Distractors:

a. is incorrect because Low-Low set valve 71D can continue to cycle.
b. is incorrect because Low-Low set valve 71D can continue to cycle.
c. is incorrect because Low-Low set valve controlled from the ASD room is not be able to cycle on Low-Low set.

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295018 AK3.03
Importance Rating	3.1

295018 Partial or Total Loss of CCW:

Knowledge of the reasons for the following responses as they apply to partial or complete loss of component cooling water:
(CFR: 41.5 / 45.6)

AK3.03 Securing individual components (prevent equipment damage)

Question: 7

The plant is operating at rated power.

- A loss of all REC pumps occurs.
- 5.2REC, LOSS of REC is entered.
- All attempts to restore REC have failed.
- The reactor is manually scrammed.

Why is the running CRD pump required to be secured?

A. Prevent motor overheating.

B. Prevent reactor water level overfill.

C. Prevent pump bearing overheating.

D. Prevent reactor vessel stratification.

Answer:

C. Prevent pump bearing overheating.

Explanation:

The REC system supplies cooling to the CRD pump bearing and oil cooler via the non-critical supply loop. REC's only safety function is to provide cooling to the Reactor Building Quadrant room Fan Coil Units for RHR pump operation. With the loss of REC cooling the running CRD pump must be secured to prevent pump damage due to overheating. While the quadrant fan coil units provide cooling for the RHR pumps, they are not credited for keeping the CRD pump motor cool for operation. The CRD pump injects water in the bottom head region of the reactor vessel which can be a part of the reason for stratification events which occur with low core flow and cool CRD water amassing in the bottom of the reactor vessel. With the loss of REC the reactor recirculation pumps are secured so core flow is low. Also with continued CRD injection during a scram, CRD is supplying RPV injection around 140 gpm which can eventually lead to overfilling the RPV. With the new reactor vessel level control system and setpoint setdown overfill events are not as likely, but can occur if the operator does not keep track of RPV level.

From COR002-19-02 OPS Reactor Equipment Cooling

5. A sustained loss of REC to the CRD pumps would cause damage to the pumps since the bearing and oil coolers are supplied by the REC non-critical equipment supply loop.

From 5.2REC Loss of REC

3. SUBSEQUENT OPERATOR ACTIONS

3.1 Record current time and date.

3.2 IF low pressure isolation occurred and initiating condition cleared, THEN restore system from isolation per Attachment 4 (Page).

3.3 IF REC HEADER PRESSURE <u>not</u> restored after completing Immediate Operator Actions, THEN perform following:

3.3.1 **SCRAM and enter Procedure 2.1.5**.

3.3.2 Stop both Reactor Recirc pumps and enter Procedure 2.4RR.

3.3.3 Stop running CRD pump.

<u>NOTE</u> – Securing all AC lube oil pumps first will cause DC lube oil pumps to start unless DC oil pump control switches are first taken to STOP and allowed to spring return to their normal positions.

3.3.4 WHEN Recirc MG Sets have stopped, THEN perform following:

3.3.4.1 Momentarily place respective control switches for both DC lube oil pumps to STOP and allow them to spring return to their normal positions (R-958-NW at 125 VDC Reactor Bldg Starter Rack).

3.3.4.2 Shut down all AC lube oil pumps.

Distracters:

.

A. This option is incorrect because REC does not provide cooling to the motor. REC is lost to the quad fan coil units so the quadrant temperatures rise but there are no restrictions for CRD pump operation based on the quadrant temperatures. If the candidate does not know that CRD pumps are not restricted from operating due to increased room temperature they would select this option. This option is plausible because the quadrant temperature rises and the CRD pump motor operating temperature also rises.

B. This option is incorrect because vessel overfill is not an issue requiring removing the CRD pump from service. With a scram present, CRD system flow into the RPV is approximately 140 gpm. This relatively cool water expands after it is injected into the RPV. With the old reactor vessel level control system it was common to overfill the RPV due to CRD injection. The new reactor vessel level control system minimizes these events but they can still happen if the operators are slow to reset the reactor scram so this option is plausible for this reason.

D. This option is incorrect because vessel stratification is not an issue requiring removing the CRD pump from service. The CRD system injects into the bottom head region via the CRD mechanisms. If core flow is low and the CRD system is injecting, the colder water can collect in the bottom head region and cause bottom head metal temperatures to be much colder than the rest of the vessel metal temperatures. The vessel is then considered stratified. Stopping CRD flow is a means of precluding stratification, if the core flow is low because the RR pumps are tripped. However, with normal reactor scram conditions, and reactor feedwater pumps controlling RPV level, the stratification can be prevented by closing CRD-V-29, charging water isolation valve. The CRD pump is tripped because of a lack of REC cooling and not because of stratification issues.

Technical Reference(s): 5.2REC Loss of REC, Rev. 16

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR002-19-02 OPS Reactor Equipment Cooling

6. Given a specific REC malfunction, determine the effect on any of the following:

   g. CRDH system

Question Source:	Bank #	_______
	Modified Bank #	_______
	New	X
Question History:	Last NRC Exam	_______
Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (5)	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295019.AA1.02
Importance Rating	3.3

295019 Partial or Total Loss of Inst. Air- Ability to operate and/or monitor the following as they apply to partial or complete loss of instrument air:

AA1.02 Instrument air system valves: Plant-Specific

Question: 8

The plant is operating near rated power.

- A leak has developed in the Augmented Radwaste Building basement air system.
- Instrument air header pressure has lowered to 75 psig.

What action is required?

A. Close IA-SOV-21, Drywell Instrument Air Supply Valve.

B. Close IA-MO-80, Non Critical Instrument Air Isolation Valve.

C. Open SA-MO-81, Service Air to Instrument Air Crosstie Valve.

D. Ensure SA-AO-PCV-609, Service Air System Isolation Valve is open.

Answer:

B. Close IA-MO-80, Non Critical Instrument Air Isolation Valve.

A leak is present in the non-critical instrument air system. This leak is large enough to lower the instrument air header pressure. Closing IA-MO-80 from the control room isolates the leak and allows the instrument air header pressure to recover.

Explanation:

5.2Air Subsequent operator actions:

4.3 IF air drying/filtering components at fault, THEN perform following:
- 4.3.1 Open SA-MO-81, SA TO IA CROSSTIE (PANEL A).
- 4.3.2 Place standby dryer and filters in service per Procedure 2.2.59.
- 4.3.3 IF necessary, THEN manually bypass any obstructed component(s).
- 4.3.4 WHEN dryer and filter flow returned to service, THEN close SA-MO-81.

5.2Air Attachment 2 Loss of Instrument Air is entered when Instrument Air pressure lowers to less than 77 psig.

4. IA PRESSURE LOSS

4.1 Close IA-MO-80, NON CRIT INSTRUMENT AIR ISOLATION (PANEL A).

4.2 IF Primary Containment inerting in-progress, THEN perform following:

CAUTION – Rupture disc on N_2 inerting line may have blown.

4.2.1 Make gaitronics announcement to evacuate Reactor Building.

4.2.2 Restrict Reactor Building access to all personnel until habitability at acceptable levels.

4.2.3 At N_2 storage tank, close N2-V-99, NITROGEN SUPPLY ROOT ISOLATION VALVE (YD-S, south of RR Airlock).

4.2.4 Contact RP to determine Reactor Building habitability.

4.2.5 Shut down Primary Containment N_2 Inerting System per Procedure 2.2.60.

4.2.6 Open N2-V-99.

4.3 Manually control hotwell level:

4.3.1 Close MC-37, FCV-17 INLET (T-882-N east of TEC pumps).

4.3.2 Close CM-11, LCV-2C NORMAL MAKEUP INLET (T-882-N, east end of TEC HXs).

4.3.3 Close MC-776, LCV-2D NORMAL DUMP INLET (T-882-N, northeast corner above FCV-17).

4.3.4 Throttle the following, as necessary, to maintain hotwell level:

4.3.4.1 CM-16, LCV-2B, SURGE MAKEUP BYPASS (T-882-N, east end of TEC HXs).

4.3.4.2 MC-36, LCV-2A, SURGE DUMP BYPASS (T-882-N, east of TEC pumps).

4.4 Ensure N_2 lineup to drywell pneumatic header:

4.4.1 PC-12, NITROGEN MAKEUP HEADER SUPPLY, open (R-903-SW, south of RR airlock).

4.4.2 IA-SOV-21, DRYWELL IA SUPPLY VLV, closed (PNL 9-3).

4.5 IF RWCU-FCV-55 was being used for RPV level control, THEN perform following:

4.5.1 Ensure RWCU pumps off.

4.5.2 Throttle open RWCU-MO-74, DEMIN SUCTION BYPASS VLV.

4.5.3 Establish Shutdown Cooling transfer to Main Condenser or Radwaste, as necessary, for RPV level control, per Procedure 2.2.69.2.

4.6 IF SA-PCV-609, SERVICE AIR SYSTEM ISOLATION, indicates open locally, THEN close (C-882, between Air Receivers, in overhead):

4.6.1 SA-14, AIR RECEIVER 1A 6" OUTLET (C-882-S, immediately west of A SA Receiver).

4.6.2 SA-15, AIR RECEIVER 1B OUTLET (C-882-S, immediately west of B SA Receiver).

Distracters:

A. This option is incorrect because IA-SOV-21, Drywell IA Supply Valve is used to supply back up IA to the inboard MSIV in the event Nitrogen to the DW is lost. It is normally shut at power. The candidate would chose this answer if he/she believe this valve to be normally open and that by shutting the valve it would isolate the leak.

C. This option is incorrect because SA-MO-81 is required to be opened by the control room operator when Air pressure lowers to 85 psig and it has been determined that a clogged instrument air dryer is the cause of lowering air pressure. The candidate would select this if he/she remembered it is an action that is identified for IA pressure below 85 psig (but does not remember the other requirement for confirmation of a clogged dryer).

D. This option is incorrect because SA-AO-PCV-609, Service Air System Isolation Valve automatically closes at >77 psig to isolate the Service Air Header from the Instrument Air Header. The procedure has an action to reopen this valve once it has been verified there is no leak in SA header. The candidate would chose this if he/she does not remember this valve automatically closes.

Technical Reference(s): 5.2Air Loss of Instrument Air, Rev. 19

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective:

7 Given a specific Plant Air system malfunction, determine the effect on any of the following:

a. Plant operation

Question Source: Bank # ______
Modified Bank # _3979_ (See attached)

	New	________
Question History:	Last NRC Exam	________
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD: 2

Original Question 3979

Which of the following is an automatic action that occurs on lowering Instrument Air pressure?

a. SA-MO-81, SA To IA Crosstie Valve opens to bypass ALL Instrument Air Dryers.

b. IA-SOV-SPV21, Drywell IA Supply Valve opens to supply backup Instrument Air to the Inboard MSIV's.

c. SA-AO-PCV-609, Service Air System Isolation Valve closes to isolate the Service Air Header from the Instrument Air Header.

d. IA-MO-80, Non Critical Instrument Air Isolation Valve closes to isolate ALL air loads except the Reactor Building Critical Air Header.

ANSWER:

c. SA-AO-PCV-609, Service Air System Isolation Valve closes to isolate the Service Air Header from the Instrument Air Header.

EXPLANATION OF ANSWER: c. Correct. SA-AO-PCV-609 closes at 77 psig to isolate the Service Air Header from the Instrument Air Header. a,b,d. Require operator action.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.01
	Importance Rating	3.5

295021 Loss of Shutdown Cooling:

Ability to determine and/or interpret the following as they apply to loss of shutdown cooling:
(CFR: 41.10 / 43.5 / 45.13)

AA2.01 Reactor water heatup/cooldown rate

Question: 9

The plant is shutdown for refueling with the following conditions:

- The reactor has been shutdown for 48 hours.
- RPV water level is being maintained 50 to 60 inches.
- RHR HX inlet temperature indicates 100ºF.

What is MINIMUM time for the reactor coolant to reach 212ºF if shutdown cooling is lost?

A. 2.1 hours

B. 2.9 hours

C. 4.9 hours

D. 13.1 hours

Answer:

A. 2.1 hours

Explanation:

Determining the Time to boil is an interpretation of heatup rate (T_{final} –$T_{initial}$)/Time
This requires knowledge of time after shutdown, initial reactor coolant temperature, and reactor coolant inventory available. Based upon the reactor being shutdown for 48 hours, initial Reactor Coolant Temperature at 100°F, and RPV level at the high level trip results in an estimated time to boil of 2.1 hours. Abnormal Procedure 2.4SDC, Shutdown Cooling Abnormal, Attachment 5 contains a family of curves based upon RPV (or cavity) water level. Using the water level at the high level trip curves and hours after shutdown (not Days after shutdown) and the given reactor

coolant temperature, the anwer can be determined. Interpolation of the family of curves is allowed.

Distracters:

B. This option is incorrect because the time to boil under the given conditions is 2.1 hours. This option is plausible if the time since shutdown is confused with 48. hours. The candidate who uses the 48 hours on the Water Level at the flange graph would select this option.

C. This option is incorrect because the time to boil under the given conditions is 2.1 hours. This option is plausible if the time since shutdown is confused with 48 days vs. hours and water level at the flange. The candidate who uses the 48 days on the Water Level at high level trip graph would select this option.

D. This option is incorrect due to time to boil under the given conditions is 2.1 hours. This option is plausible if the time since shutdown is confused with 48 hours with water level flooded to 1001'. The candidate who uses the 48 hours on the Water Level to Level Flooded to 1001' would select this option.

Technical Reference(s): 2.4SDC, Shutdown Cooling Abnormal, Rev 14

Proposed references to be provided to applicants during examination: 2.4SDC Attachment 5

Learning Objective: INT0231002001170A Give a set of plant conditions and time of the reactor shutdown, determine: Time to core boiling.

Question Source: Bank # ______
Modified Bank # 9181 (See attached)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (14)

Comments:

LOD 3

QUESTION: 269 9181 (1 point(s))

The plant is shutdown for refueling with the following conditions:

- ☐ The reactor has been shutdown for 48 hours.
- ☐ The steam separator is being removed.
- ☐ Flood up of the reactor cavity has begun.
- ☐ RHR HX inlet temperature indicates 105°F as read on RHR-TI-131.
- ☐ RPV level just reaches the vessel flange when an instrument malfunction results in a shutdown cooling high pressure isolation.

If cooling cannot be re-established, what is the MINIMUM time for the reactor coolant to reach 212°F?

a. 2.9 hours

b. 4.8 hours

c. 7.5 hours

d. 12.5 hours

ANSWER: 269 9181

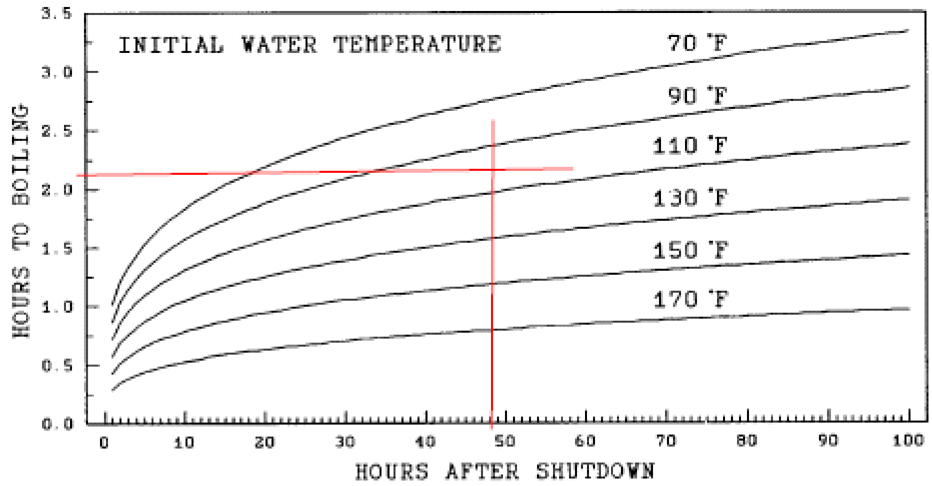
a. 2.9 hours

based on time to boiling curve Figure 2.

b. based on 48 days shutdown & water level at high level trip.
c. based on 48 days shutdown.
d. based on level at 1001 ft.

REFERENCE TO BE PROVIDED TO THE STUDENTS: [illegible]

ATTACHMENT 5 TIME TO CORE BOILING/TIME TO CORE UNCOVERY



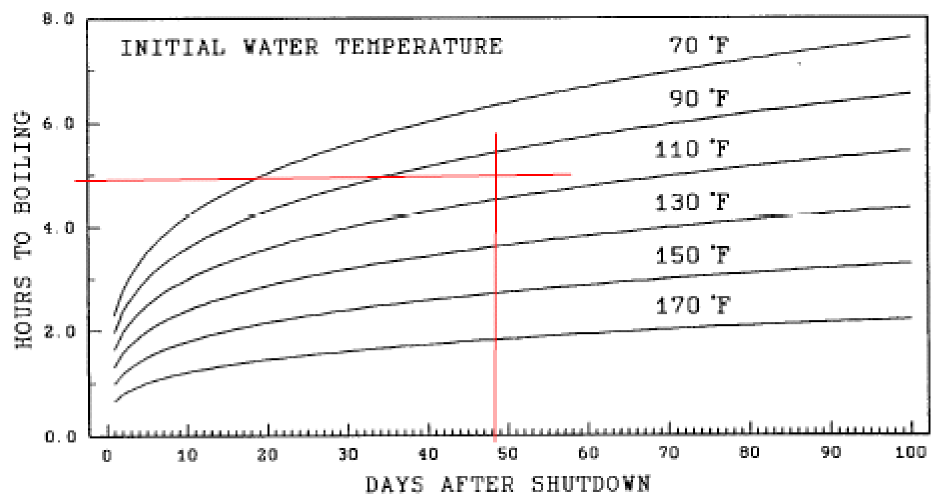
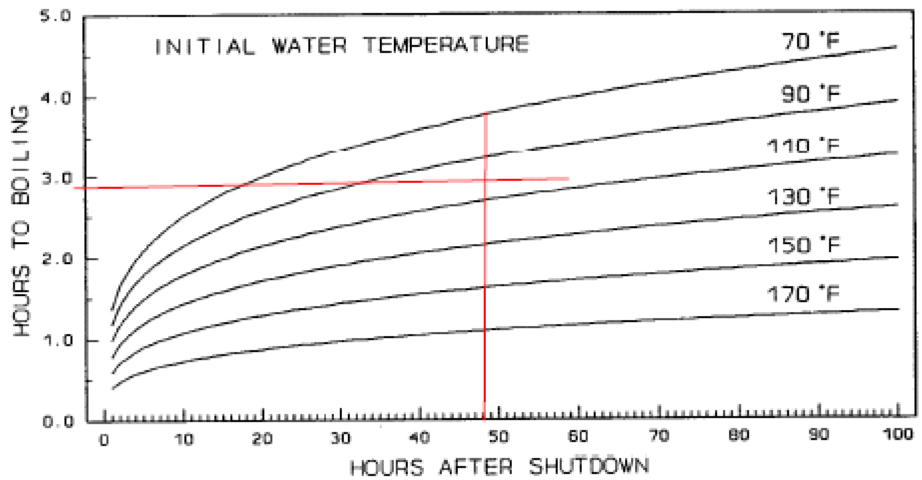


Figure 1 - TIME TO BOILING - WATER LEVEL AT HIGH LEVEL TRIP



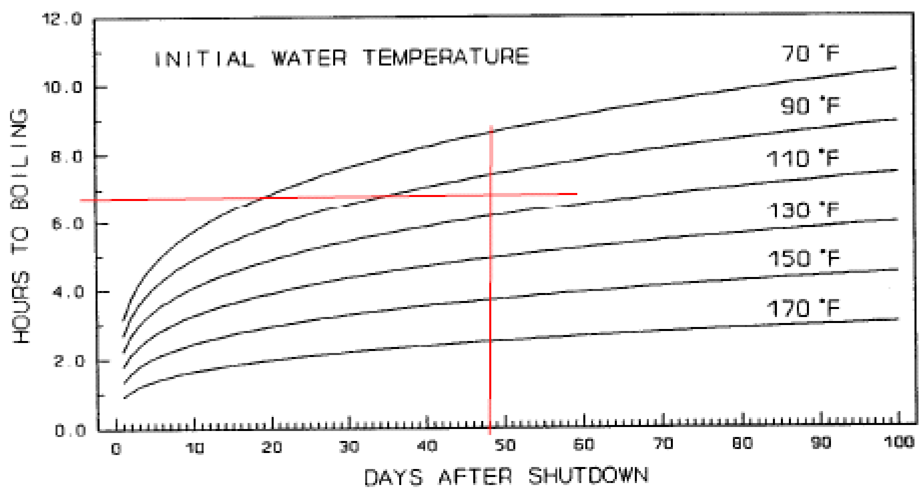
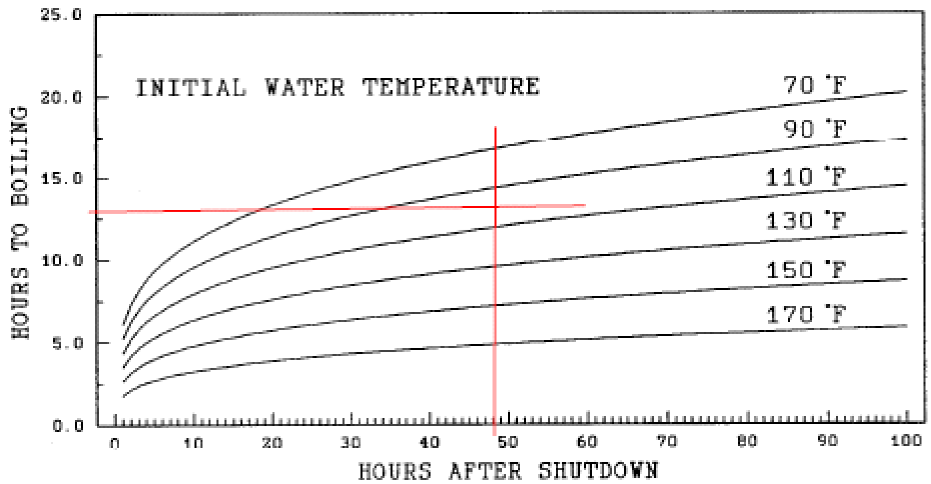


Figure 2 - TIME TO BOILING - WATER LEVEL AT FLANGE



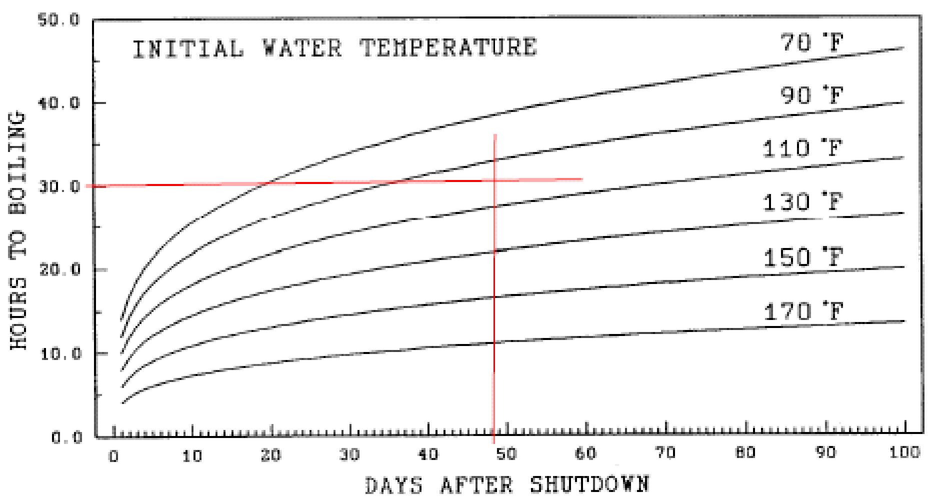


Figure 3 - TIME TO BOILING - WATER TO LEVEL FLOODED TO 1001'

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 2.2.40
	Importance Rating	3.4

295023 Refueling Accidents:

2.2.40 Ability to apply Technical Specifications for a system.
(CFR: 41.10 / 43.2 / 43.5 / 45.3)

Question: 10

The plant is in day 12 of a refueling outage.

- Refueling operations are in progress.
- A control rod is accidentally dropped into the RPV.

What is the required MINIMUM Technical Specifications water level above the top of the RPV flange to ensure off site dose is maintained within the allowable limits?

A. 6 feet

B. 12 feet

C. 21 feet

D. 37 feet

Answer:

C. 21 feet

Explanation:

From TS 3.9.6

3.9 REFUELING OPERATIONS
3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV **water level shall be ≥21 ft above the top of the RPV flange.**

APPLICABILITY: During movement of irradiated fuel assemblies with in the RPV, during movement of new fuel assemblies or handling of control rods within the RPV, when irradiate fuel assemblies are seated within the RPV.

From TS Basis

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a refueling accident in containment postulated by Reference 1. A minimum water level of 21 ft allows a decontamination factor of 200 to be used in the accident analysis for halogens (Ref. 1). This relates to the assumption that 99.5% of the total halogens released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water. The fuel pellet to cladding gap activity assumes RG 1.183, Table 3 non-loss-of-coolant-accident gap fractions (Ref. 5).

Distracters:

A. This option is incorrect because the listed water level is too low. The required level is a minimum of 21 feet above the flange. This option is plausible because it is the minimum water level above a suspended fuel bundle and the candidate may recall this number and select this option.

B. This option is incorrect because the listed water level is too low. The required level is a minimum of 21 feet above the flange. This option is plausible because it is the height of a fuel bundle or a control rod and the candidate may recall this number and select this option.

D. This option is incorrect because the listed water level is too high. The required level is a minimum of 21 feet above the flange. This option is plausible because it is the normal spent fuel water level and the candidate may recall this number and select this option.

Technical Reference(s): Technical Specifications 3.9.6

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: INT007-05-10 OPS CNS Tech Specs 3.9, Refueling Operations

1. Given a set of plant conditions, recognize non-compliance with a Section 3.9 LCO.

Question Source: Bank # _____
Modified Bank # _______
New ___X___

Question History: Last NRC Exam _______

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 2

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295024EK1.01
Importance Rating	4.1

295024 High Drywell Pressure- Knowledge of the operational implications of the following concepts as they apply to high drywell pressure:

EK1.01 Drywell integrity: Plant-Specific

Question: 11

During a LOCA the following conditions exist:

- Average drywell temperature is 210°F and steady.
- Drywell pressure is 19 psig and rising.
- Average suppression pool temperature is 205°F and rising.
- Torus water level is 14 feet and rising slowly.
- Reactor pressure is 1000 psig and steady.
- Reactor water level +25 inches (narrow range) and steady.

Why is Emergency Depressurization performed at this time?

To prevent…

A. chugging, and possible loss of the pressure suppression function.

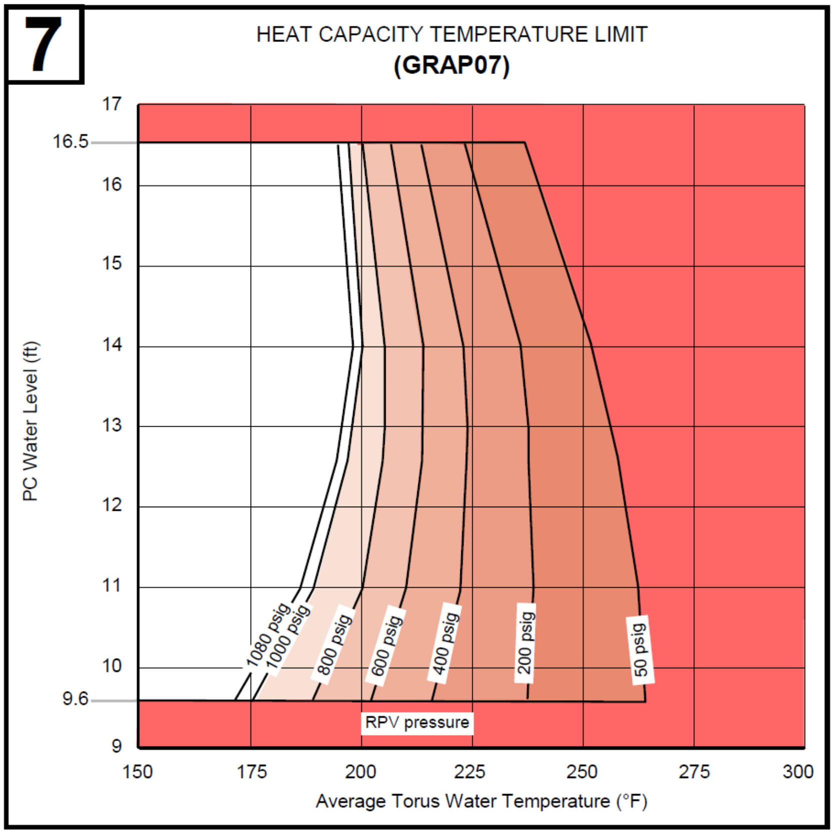
B. raising torus pressure above the PCPL A which may result in failure of containment.

C. excessive torus to drywell vacuum breaker operation and possible vacuum breaker failure.

D. the torus water level rise which may cause loss of containment on SRV actuation due to a water column in the system discharge piping.

Answer:
B. raising torus pressure above the PCPL A which may result in failure of containment.

Explanation:
Operation is on the wrong side of the Heat Capacity Temperature limit curve for the given reactor pressure. Emergency depressurizing too far above this point has the potential to lead to exceeding the PCPL A limit which could result in loss of containment integrity.



Current conditions place the plant on the unsafe side of the HCTL Curve.

A. Heat Capacity Temperature Limit (HCTL) (GRAP07)

1. Definition - **the highest torus water temperature from which emergency RPV depressurization will not raise suppression chamber pressure above Primary Containment Pressure Limit A, while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.**

2. Use - The HCTL is a function of RPV pressure and primary containment water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant.

   Flowchart 3A requires emergency RPV depressurization when RPV pressure and torus water temperature can not be maintained within HCTL.

B. Primary Containment Pressure Limits A/B (GRAP11)

1. Definition - the lesser of either:

    a. **The pressure capability of the primary containment**, or

    b. The maximum primary containment pressure at which vent valves sized to reject all decay heat from the containment can be opened and closed.

    c. For Primary Containment Pressure Limit A, the maximum primary containment pressure at which SRVs can be opened and will remain open.

2. Use - Each PCPL is a function of primary containment water level and primary containment pressure. The limits are utilized to avoid challenges to primary containment vent valve operability, SRV operability, and primary containment integrity.

RPV vent valve operability is not a concern in derivation of the PCPL because RPV venting can be accomplished using the motor operated main steam line drain valves. Operability of these valves are not affected by containment atmospheric pressure.

Distracters:

A. This option is incorrect because chugging is not an issue with emergency depressurizing. Chugging is an issue when drywell sprays are initiated when suppression chamber pressure exceeds the Suppression Chamber Spray Initiation Pressure (SCSIP) to preclude chugging—the cyclic condensation of steam at the downcomer openings of the drywell vents. When a steam bubble collapses at the exit of the downcomers, the rush of water drawn into the downcomers to fill the void induces stresses at the junction of the downcomers and the vent header. Repeated application of such stresses could cause fatigue failure of these joints, thereby creating a direct path between the drywell and suppression chamber. Steam discharged through the downcomers could then bypass the suppression pool and directly pressurize the primary containment. This option is plausible because the chugging phenomenon is a real consideration in certain conditions and this is a common misconception among operators.

C. This option is incorrect because excessive torus to drywell vacuum breaker operation is not occurring so valve failure is not an issue. This option is plausible because all the energy added to the suppression pool causes suppression pool level and air space pressure to rise. Vacuum breaker operation may occur but not excessively. If the candidate believes adding the energy to the suppression pool at the given conditions may effect vacuum breaker integrity would select this option.

D. This option is incorrect because the primary containment water level of 16 ft is defined as the SRV Tail Pipe Level Limit (SRVTPLL). It is the highest primary containment water level at which opening of an SRV will not result in exceeding the capability of the SRV tail pipe, tail pipe supports, T quencher, or T quencher supports. By maintaining primary containment water level below this limit, SRV system damage and containment failure may be precluded.

This option is plausible because had the torus level been higher than the conditions specified in the stem then this would have been the correct answer. So a candidate who does not recall the SRVTPLL may very well choose this option.

Technical Reference(s): INT0080613, OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL, Rev 17
AMT-TB00 Appendix B (PSTG) for Step RC/P-2, Rev 8

Proposed references to be provided to applicants during examination: _HCTL Graph

Learning Objective:

4. State the basis for primary containment control actions as they apply to the following.

    a. Specific setpoints
    b. Primary Containment Control Systems
    c. Graphs referenced on Flowchart 3A

Question Source:	Bank #	_21460______
	Modified Bank #	_______
	New	_______
Question History:	Last NRC Exam	________
Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 (8)	

Comments:

LOD: 4

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295025.EK2.07
Importance Rating	3.7

295025 High Reactor Pressure- Knowledge of the interrelations between high reactor pressure and the following:

EK2.07 RCIC: Plant-Specific

Question: 12

Given the following conditions:

- Reactor Pressure is 600 psig.
- Reactor Core Isolation Cooling (RCIC) is operating for level control.
- The flow controller is in MANUAL.
- RCIC injection flow is 375 gpm.

Ten minutes later Reactor Pressure is 1070 psig and rising.

What effect, if any, does the rise in reactor pressure have on RCIC turbine speed and injection flow?

	RCIC turbine speed	RCIC injection flow
A.	lowers	lowers
B.	unaffected	lowers
C.	rises	rises
D.	unaffected	rises

Answer:

B. unaffected lowers

Explanation:

The RCIC-FIC-91, FLOW CONTROLLER, operates in manual and automatic modes. With the RCIC controller in manual, RCIC's EGR receives a set speed signal from the controller. Transient changes in pressure may initially act to cause the turbine speed to change, however the EGR will correct those and maintain the turbine at the set speed. As reactor pressure rises, pump speed would remain the same, resulting in lower pump flow.

Distracters:

A. This option is incorrect because even though the RCIC injection would lower the speed would remain constant as it is controlled at a constant speed with the controller in manual. A candidate who believes that the turbine throttle valve is held in a specific position when RCIC is in manual and who also believes that higher reactor pressure causes higher required pumping power may choose this option.

C. This option is incorrect because the turbine speed remains steady and system flow lowers. If the candidate does not understand that with the RCIC controller in manual RCIC turbine speed is controlled based on the controller setting, he/she may select this option. With no operator action the turbine speed will not change. This option is plausible because a rise in reactor pressure provides a larger motive force to speed up the turbine and could raise system flow.

D. This option is incorrect because the system flow lowers. If the candidate does not understand that with no change in the turbine speed (thus pump speed) the flow from the pump will be lower when discharging to the higher reactor pressure, he/she may select this option. This option is plausible because the turbine speed is correct.

Technical Reference(s): Vendor Manual 1869
COR0021802 OPS Reactor Core Isolation Cooling, Rev. 25

Proposed references to be provided to applicants during examination: NONE

Learning Objective: COR0021802 OPS Reactor Core Isolation Cooling

10. Predict the consequences of the following on the RCIC system:

a. High/Low Reactor Pressure

Question Source:	Bank #	3760
	Modified Bank #	____
	New	____
Question History:	Last NRC Exam	____
Question Cognitive Level:	Memory or Fundamental Knowledge	____
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295026EK3.04
Importance Rating	3.6

295026 Suppression Pool High Water Temp- Knowledge of the reasons for the following responses as they apply to suppression pool high water temperature:

EK3.04 SBLC injection

Question: 13

An Anticipated Transient Without Scram (ATWS) occurs.

- Average Reactor Thermal Power is 15% and steady.
- Suppression Pool temperature is 96° F **AND** rising.

What is the reason boron injection must be initiated before average suppression pool water temperature reaches 140° F?

A. Prevents exceeding the 25% peak-to-peak periodic neutron flux oscillations.

B. Prevents violating Technical Specification Limit for Suppression Pool Temperature.

C. Ensures the reactor will be shutdown under all conditions before the suppression pool is heated beyond its design limits.

D. Ensures the reactor will be shutdown under hot-standby conditions before the suppression pool reaches the Heat Capacity Temperature Limit.

Answer:

D. Ensures the reactor will be shutdown under hot-standby conditions before the suppression pool reaches the Heat Capacity Temperature Limit.

Explanation:

The BIIT for 15% power is 140° F. Boron injection before this temperature ensures the Reactor will be shutdown before the HCTL is exceeded.

The Boron Injection Initiation Temperature (BIIT) is the greater of:

- The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

- The suppression pool temperature at which a reactor scram is required by Technical Specifications.

The BIIT is a function of reactor power. If boron injection is initiated before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heatup rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event. Refer to Section 16 of this appendix for a detailed discussion of the BIIT.
Distracters:

A. This option is incorrect because boron is injected after large neutron oscillations are observed not before. This option is plausible because boron injection is directly tied to large neutron oscillations and boron is injected to minimize them. The candidate who does not know the boron injection is performed in response to power oscillations would select this option.

B. This option is incorrect because the Technical Specification limit for suppression pool temperature is already violated. This option is plausible because there is a Technical Specification limit on pool temperature. The candidate who recalls the Technical Specification limit and not the BIIT limit would select this option.

C. This option is incorrect because the suppression pool design temperature limit is not tied to the start of boron injection bases upon pool temperature. Getting the reactor shutdown before HCTL is reached ensures the pool can absorb the energy released to the pool during emergency depressurization. This option is plausible because shutting the reactor down will lessen the energy being transferred to the suppression pool. The candidate who cannot recall the reasons for the BIIT curve would select this option.

Technical Reference(s): AMP-TB00 (CNS PSTGs) Rev. 8 Appendix B

Proposed references to be provided to applicants during examination: EOP Graph 8 (BIIT)

Learning Objective: INT008-06-06. OPS EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram)

3. Describe the conditions that require boron injection, and when boron injection can be secured.

INT008-06-18, OPS EOP and SAG Graphs and Cautions

1. Using the graphs provided in the EOP and SAG Graphs Flowchart, explain how the shape of each curve or family of curves was determined.

Question Source:	Bank #	5334
	Modified Bank #	
	New	
Question History:	Last NRC Exam	____________
Question Cognitive Level:	Memory or Fundamental Knowledge	X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (5)

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295028EA1.01
Importance Rating	3.8

295028 High Drywell Temperature- Ability to operate and/or monitor the following as they apply to high drywell temperature:

EA1.01 Drywell spray: Mark-I&II

Question: 14

A steam leak in containment has resulted in the following conditions:

- Drywell pressure is 10 psig rising 1 psig/min.
- Torus pressure is 8 psig rising 1 psig/min.
- Drywell average temperature is 275°F and rising 5°F/min.
- All DW FCUs unavailable.
- REC is NOT available to the drywell.

What action(s) is/are required?

A. Spray the Drywell only.

B. Emergency depressurize the RPV.

C. Emergency vent the containment.

D. Spray both the Torus and the drywell.

Answer:

D. Spray both the Torus and the drywell.

Explanation:

With drywell temperature rising and no FCUs or REC available drywell temperature continues to rise. EOP-3A directs drywell sprays BEFORE average drywell temperature reaches 280°F which is the drywell design temperature. With the drywell temperature rise rate given, the drywell design temperature is reached in 1 minute. This leg does not require initiating Torus sprays prior to DW Sprays. EOP-3A directs torus sprays BEFORE torus pressure reaches 10 psig. Per EOP-3A pressure leg, torus sprays are required. Per EOP-3A drywell temperature leg, drywell sprays are required.

Distracters:

A. This answer is incorrect because torus sprays are also required to be initiated. If the candidate uses the DW temperature leg of EOP-3A only and does not also consider the containment pressure leg, he/she would select this answer. This is a common error.

B. This answer is incorrect because emergency depressurization is not required until temperature cannot be restored and maintain less than 340°F. The Candidate may select this answer if he/she recalls the containment design temperature is 281°F and believes sprays should be initiated for this reason. If DW temperature exceeds 340°F, and drywell sprays are placed in service and lowers temperature less than 340°F then emergency depressurization would not be appropriate. The key EOP verbage is the drywell temperature "cannot be restored and maintained below….".

C. This answer is incorrect because emergency venting containment is only allowed to prevent exceeding PCPL-A. Before emergency venting is allowed, emergency depressurization is required. There are no parameters driving emergency depressurization or emergency venting at this time. This answer is plausible because the drywell is emergency vented under more severe circumstances. The candidate who believes PCPL-A is threatened would select this answer.

Technical Reference(s): EOP-3A Primary Containment Control, Rev. 15
2.2.69.3 RHR Suppression Pool Cooling and Containment Sprays, Rev. 46

Proposed references to be provided to applicants during examination: EOP Graph 9_DWSIL
EOP 3A Temperature and Pressure partials.

Learning Objective: INT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

11. Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Question Source: Bank # ______
Modified Bank # 19310 (See attached)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Original Bank Question:

A steam leak in containment has resulted in the following conditions:

- Drywell pressure is 6 psig rising.
- Torus pressure is 5 psig rising.
- Drywell average temperature is 200°F rising.
- All Drywell FCUs are operating.
- REC is not available.

What is the potential impact on primary containment and what action is required?

a. drywell design temperature may be exceeded; initiate drywell sprays **only**.

b. drywell design temperature may be exceeded; initiate torus **and** drywell sprays.

c. ADS valves environmental qualifications may be exceeded; initiate torus sprays **only.**

d. ADS valves environmental qualifications may be exceeded; emergency depressurize the reactor.

ANSWER: 95 19310

b. drywell design temperature may be exceeded; initiate torus **and** drywell sprays.

With drywell temperature rising and no REC available then FCUs will not turn drywell temperature. 3A directs drywell sprays before average drywell temperature reaches 280°F which is the drywell design temperature. Torus sprays should be initiated prior to initiating drywell sprays.

a. Initiating drywell sprays only is not allowed per 2.2.69.3 though this could be read directly off the temperature leg of 3A.
c. Temperature is approaching 280°F which is our drywell design temperature which is lower than the ADS valve environmental qualifications. Torus sprays are allowed however implementing tours sprays alone will not reduce drywell temperature with a steam leak in the drywell.
d. Emergency depressurization is not required until temperature cannot be restored and maintain less than 280°F.

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295030EA2.02
Importance Rating	3.8

295030 Low Suppression Pool Water Level- Ability to determine and/or interpret the following as they apply to low suppression pool water level:

EA2.02 Suppression pool temperature

Question: 15

Following a LOCA the following conditions are present:

- Only RHR pump C is injecting.
- Torus pressure is 4.5 psig (stable).
- Torus average water temperature is 175°F (rising slowly).
- Primary containment water level is 5.5 feet (stable).

The CRS has directed to BOP to comply with NPSH and Vortex limits.

What is the MAXIMUM flow allowed for RHR pump C?

A. 6500 gpm

B. 7000 gpm

C. 7300 gpm

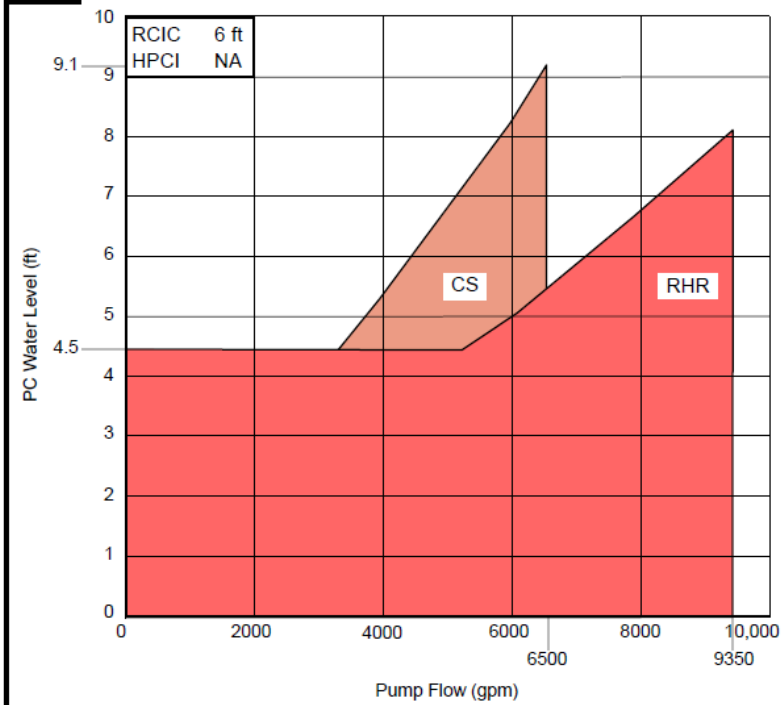
D. 7500 gpm

Answer:
A. 6500 gpm

4

VORTEX LIMITS

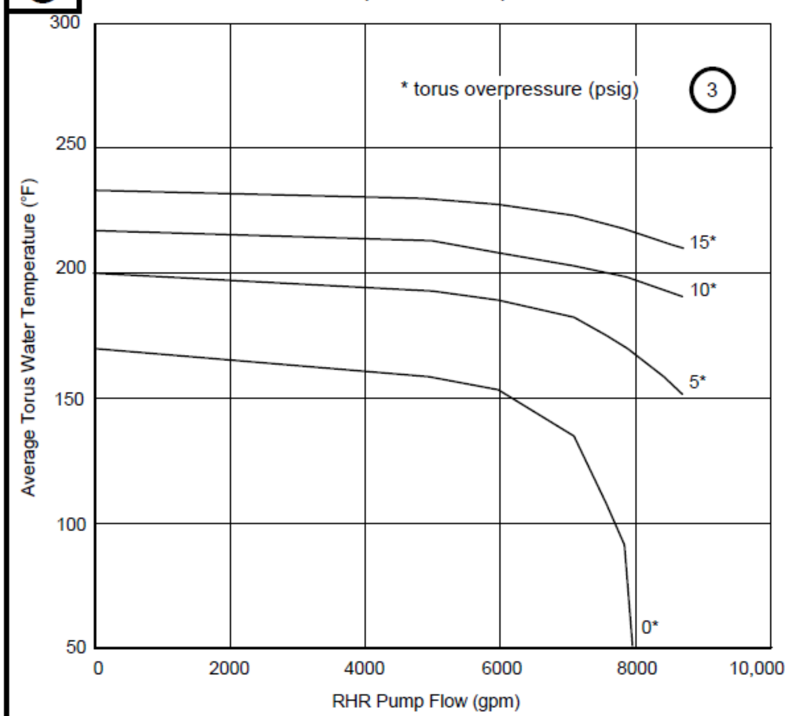
(GRAP4A, B 6A, B)



5

RHR PUMP NPSH LIMIT

(GRAP5A, B)



NOTE 3
Torus overpressure is sum of torus pressure and hydrostatic head above suction strainer

Torus pressure (psig)			____________
Hydrostatic head (psig)			
PC water level (ft.)	____________		
Strainer level (ft.)	-4		
0.43 x	____________	=	+ ____________
Torus overpressure (psig)			____________

Explanation:

With 4.5 psig pressure and 5.5 feet of water level in the torus, there is 4.43 psig overpressure in the torus. The NPSH flow limit is approximately 7400 gpm. At 5.5 feet of torus water level, the vortex limit for RHR is 6500 gpm. With RPV level rising and above the top of active fuel, the CRS is correct in limiting flow to the NPSH and vortex limits. The maximum RHR flow is 6500 gpm. When using EOP graphs, Procedure 5.8, EMERGENCY OPERATING PROCEDURES, (Step 3.11) allows interpolation of graphs if SPDS is unavailable. SPDS is unavailable because the curves and graphs are given. Step 3.11 also states interpolation is not required even though it is preferred. The question is asking for the maximum allowed flow so interpolation is required in this question.

Distracters:

B. This option is incorrect because the listed flow is too high. This is a flow that is below the NPSH limit and the candidate may believe it is best to be conservative with the flow for NPSH conservatism. The vortex graph is more limiting. The candidate who believes the NPSH graph is the most limiting would select this option. This option is plausible because it is conservative for the NPSH limit.

C. This option is incorrect because the listed flow is too high. This is the limiting flow if using the interpolated NPSH graph. The vortex graph is more limiting. The candidate who believes the NPSH graph is the most limiting would select this option. This option is plausible because it is the correct NPSH limit.

D. This option is incorrect because the listed flow is too high. If the candidate uses the 5 psig curve for determining NPSH limits this would be the answer. For conservatism, when using curves, the most conservative one is used if interpolation is not used. This option is plausible because it is the answer one would get if using the incorrect curve.

Technical Reference(s): CNS PSTG AMP TB00, Section 16, Rev. 8
Proposed references to be provided to applicants during examination: EOP Graphs 4 and 5, EOP Note 3

Learning Objective:
INT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

4. State the basis for primary containment control actions as they apply to the following.

c. Graphs referenced on Flowchart 3A

Question Source:	Bank #	16569	
	Modified Bank #	________	
	New	________	
Question History:	Last NRC Exam	2003	
Question Cognitive Level:	Memory or Fundamental Knowledge		______
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41 (10)		

Comments:

LOD: 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 2.4.2
	Importance Rating	3.8

295031 Reactor Low Water Level:

2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Question: 16

RPV water level lowers to the value that is an entry condition for EOP-1A, RPV CONTROL and remains steady at that level.

No other EOP entry conditions are satisfied.

What group isolation(s) automatically occur(s)?

A. Group 2 **only**.

B. Group 3 **only**.

C. Groups 3 and 6 **only**.

D. Groups 2, 3, and 6.

Answer:

A. Group 2 **only**.

Explanation:

Instrument zero RPV level at CNS is +165 inches above the top of active fuel. Normal RPV water level at CNS is +35 inches above instrument zero. EOP-1A entry condition on low RPV level is +3 inches above instrument zero. The PCIS Group 2 isolation actuates at +3 inches RPV level. The PCIS Group 3 and 6 isolations both actuate at -42 inches wide range which is 45 inches RPV level below the Group 2 actuation point.

Distracters:

B. This option is incorrect because a Group 3 isolation does not occur until RPV level is at -42 inches. The candidate who does not recall this setpoint would select this option This option is plausible because Group 3 does occur on a low RPV level condition.

C. This option is incorrect because the Group 3 & 6 isolations do not occur until the -42 inches level is reached. The candidate who does not recall this setpoint would select this option. This option is plausible because the Groups 3 & 6 isolation do occur on a low RPV level.

D. This option is incorrect because the Group 3 and 6 isolations do not occur until -42 inches RPV level. The candidate who does not recall these Group isolations actuation points would select this option. This option is plausible because the group 2, 3 and 6 isolations occur on a low RPV level condition.

Technical Reference(s): 2.1.22 Recovering From A Group Isolation, Rev. 59

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR0020302 OPS CONTAINMENT

21. Given plant conditions, determine if the following should have occurred:
    b. Any of the PCIS group isolations

Question Source:	Bank #	_19694_
	Modified Bank #	________
	New	________
Question History:	Last NRC Exam	_______
Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK1.05
	Importance Rating	3.4

295037 SCRAM Condition Present and Reactor Power above APRM Downscale or Unknown:

Knowledge of the operational implications of the following concepts as they apply to scram condition present and reactor power above APRM downscale or unknown: (CFR: 41.8 to 41.10)

EK1.05 Cold shutdown boron weight: Plant-Specific

Question: 17

The plant is operating at full power when the following occurs:

- All MSIVs close resulting in a Reactor Scram with 27 rods failing to fully insert.
- RPV water level is intentionally lowered due to level/power conditions being met.
- SLC is injecting Boron.

When can a normal reactor depressurization commence?

A. When Hot Shutdown Boron Weight is injected.

B. When Cold Shutdown Boron Weight is injected.

C. When all APRMs indicate less than 3% power (downscale).

D. When all but one control rod is inserted to position 02.

Answer:

B. When Cold Shutdown Boron Weight is injected.

Explanation:

When the reactor is not shutdown and lowering RPV level to lower reactor power is required, boron is being injected into the RPV. When hot shutdown boron weight (26% of the Standby Liquid Control tank level) has been injected into the RPV, the RPV level is raised via outside the shroud injection systems. RPV level is raised to bring more of the boron into fuel region to aid in keeping reactor power low. The normal RPV level band of +3 inches to +54 inches is directed at this point. Once cold shutdown boron weight is injected (60% of the Standby Liquid Control tank level) or it is determined the reactor will remain shutdown without relying on the boron concentration in the RPV, reactor pressure can be lowered so the plant can be placed in a cold shutdown condition.

From training material INT0080606 OPS EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram):

FS/Q-16 - Injection normally continues until the entire usable contents of the SLC tank have been injected. Actions in other EOP steps, however, are conditioned upon lesser amounts of boron:

a. When the Hot Shutdown Boron Weight has been injected, RPV water level may be restored above the low level scram setpoint. The second override in Flowchart 7A Step FS/L-11 becomes active.

b. When the Cold Shutdown Boron Weight has been injected, RPV cooldown may be initiated in accordance with Step FS/P-6.

FS/P-6 - RPV depressurization and cooldown may not proceed until the reactor is shutdown with no boron injected or the amount of boron injected into the RPV is sufficient to keep the reactor shut down.

A volume of boron solution equivalent to 60% of the SLC tank (or 2258 lbs of boric acid and 2321 lbs of borax) is called the Cold Shutdown Boron Weight (CSBW). The CSBW is defined to be the amount of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions.

The CSBW is determined assuming:

a. All rods are full out
b. Core is at most reactive exposure
c. No Xenon in the core
d. No voids in the core
e. Water is at most reactive temperature (68˚F)
f. Shutdown cooling and RWCU are in service
g. RPV water level at 54 in.

NOTE 1 of EOP 6A provides a method of calculating the SLC tank level resulting from injection of the CSBW. Weights of borax and boric acid are provided if boron injection must be performed by preparing the borate solution at the RWCU precoat tank.

If any amount of boron less than the CSBW has been injected into the RPV, cooldown is not permitted unless it can be determined that control rod insertion alone assures the reactor will remain shutdown under all conditions. The core reactivity response from cooldown in a partially borated core is unpredictable and subsequent steps may not prescribe the correct actions for such conditions if criticality were to occur.

The reactor is shutdown under all conditions without boron if all control rods are inserted to or beyond position 02, the shutdown margin is met (theoretically strongest control rod withdrawn and all other control rods fully inserted), or engineering determination.

Distracters:

A. This option is incorrect because HSBW only allows water level to be restored to +3 inches to +54 inches to promote boron mixing in the core. If the candidate does not understand that cold shutdown boron weight (CSBW) must be injected prior to cooling down the reactor, he/she would select this option. This option is plausible because HSBW does allow a major parameter change to be made during ATWS conditions.

C. This option is incorrect because APRM downscales are not used to determine when the cooldown can commence. If the candidate does not understand that cold shutdown boron weight (CSBW) must be injected prior to cooling down the reactor, he/she would select this option. This option is plausible because APRM downscale indication is used to determine if EOPs are entered.

D. This option is incorrect because with 1 control rod not at 02 and all other control rods at 02, the reactor can not be considered shut down under all conditions without boron. The RE would be required to evaluate this and inform the control room. If the candidate thinks he/she can make this call without RE evaluation they would select this option. This option is plausible because 1 control rod beyond position 02 and all other control rods at position 00 is the definition of shutdown margin and the reactor can be considered shutdown at this point and the cooldown could commence.

Technical Reference(s): EOP-6A, Rev 16/EOP-7A, Rev 17-Failure to Scram EOPs

Proposed references to be provided to applicants during examination: NONE

Learning Objective: INT0080606 OPS EOP Flowchart 6A - RPV Pressure & Power (Failure-to-Scram)

12. Given an EOP flowchart 6A, RPV PRESSURE/POWER step, state the reason for the actions contained in the step.

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK2.02
	Importance Rating	3.6

295038 High Off-site Release Rate:

Knowledge of the interrelations between high off-site release rate and the following: (CFR: 41.7 / 45.8)

EK2.02 Offgas system

Question: 18

The plant is operating at near rated power when significant fuel failures occur.

- EOP-5A is entered.
- Elevated Release Point (ERP) radiation rates are trending up on PMIS.
- Standby Gas Treatment A is operating in support of an upcoming HPCI surveillance.

What operation lowers the Offsite release rate?

A. Secure Standby Gas Treatment Train.

B. Secure the Mechanical Vacuum pump.

C. Ensure OG-254, OFFGAS SYSTEM ISOLATION is open and AOG-902, AOG RETURN is closed.

D. Ensure both OG-254, OFFGAS SYSTEM ISOLATION and AOG-902, AOG RETURN are closed.

Answer:

D. Ensure both OG-254, OFFGAS SYSTEM ISOLATION and AOG-902, AOG RETURN are closed.

Explanation:

When at this power level, the Off Gas and Augmented Off Gas systems are in service. The Off Gas system discharges out the ERP. The Off Gas system transports the stream through the AOG system and then out the ERP. The mechanical vacuum pump discharges out the ERP when it is running. The Standby Gas Treatment system draws from secondary containment and the HPCI gland seal exhauster and discharges out through the ERP. When off-gas isolates on high radiation, OG-254 receives a closed signal and AOG-902 receives a closed signal.

Distracters:

A. This answer is incorrect because the Standby Gas Treatment system is not drawing gland steam from HPCI at this point. If HPCI were running, then noble gases would be drawn by the SGT train and discharged through the ERP. The candidate who believes SGT is exhausting the noble gases from the HPCI system would believe securing SGT would lower the off-site release rate. This answer is plausible because the SGT system does discharge through the ERP.

B. This answer is incorrect because the Mechanical Vacuum pump is not running at this power level. If the MVP were running with the given conditions, then the off-site release rate would be lower as the MVP discharges, unfiltered, directly through the ERP. This answer is plausible because the MVP does discharge through the ERP.

C. This answer is incorrect because OG-254 should be closed and not opened. The action to open OG-254 and close AOG-902 are the actions to take if AOG does not isolate when required. The candidate who confuses OG and AOG isolations may select this answer. This answer is plausible because it is the actions taken if AOG fails to isolate.

Technical Reference(s): SOP 2.2.62 Off Gas System, Rev. 27

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR001-161-1 OPS Off Gas, Rev 27

11. Given an Off Gas system component manipulation, predict and explain the change in the following parameters:

b. Radioactive release rate

Question Source:	Bank #	______
	Modified Bank #	______
	New	___X___
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	__ __
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000AK3.04
	Importance Rating	2.8

600000 Plant Fire On Site

Knowledge of the reasons for the following responses as they apply to plant fire on site:

AK3.04 Actions contained in the abnormal procedure for plant fire on site

Question: 19

The plant is operating at 100% when the Shift Manager directs a Control Room evacuation due to a fire in the control room.

Per Emergency Procedure 5.4FIRE-S/D, FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM, why are all the AC powered reactor feedwater pump lube oil pump control switches placed in PULL-TO-LOCK?

A. To ensure a reactor water overfill event is prevented.

B. To ensure reactor water level is intentionally lowered to aid in FW preheating.

C. To ensure the RFPs are operating on the DC lube oil pumps which maintain bearing lubrication during pump operation.

D. To ensure the RFPs are operating on the DC lube oil pumps which maintain bearing lubrication during pump coast down.

Answer:

A. To ensure a reactor water overfill event is prevented.

Explanation:

Prior to evacuation of the Control Room, the RFLO pumps are placed in PULL-TO-LOCK to prevent overfill event from occurring at ~ 50 seconds following the scram. This action results in the RFPs tripping and it is expected that Reactor water level lowers and HPCI and RCIC are used for level control.

Distracters:

B. This answer is incorrect because the intent of the procedure guidance is to prevent an overfill event. RPV level is intentionally lowered during ATWS events to preheat the feedwater and lower reactor power which makes this answer plausible. The candidate who recalls intentionally lowering RPV level by tripping the RFPs would select this answer.

C. This answer is incorrect because the intent of the procedure guidance is to prevent an overfill event. This option starts the DC lube oil pumps for the RFPs but these pumps are only designed to protect the RFPs during coast down and not sufficient for the pump to continue to operate and automatically control level. The candidate selects this if it is believed the RFPs can continue to be controlled by RVLC in automatic with only the DC lube oil pumps. This answer is plausible because the DC oil pumps are provided to ensure proper lubrication.

D. This answer is incorrect because the intent of the procedure guidance is to prevent an overfill event. This will ensure the DC lube oil pumps are running to protect the RFPs during coast down, but this is not the reason the AC lube oil pumps are placed in PTL. The candidate would select this if it is understood that later in the procedure AC power is removed from these pumps.

Technical Reference(s): 5.4 FIRE-SD FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM, Rev. 62.

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: INT032-01-34 OPS CNS Abnormal Procedures (RO) Fire

H. Given plant condition(s) and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #	_______
	Modified Bank #	_______
	New	___X___
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AA1.01
	Importance Rating	3.6

700000 Generator Voltage and Electric Grid Disturbances:

Ability to operate and/or monitor the following as they apply to generator voltage and electric grid disturbances:

AA1.01 Grid frequency and voltage

Question: 20

The plant is operating at 70% power when Doniphan Control Center (DCC) System Operator notifies the Control Room that a grid disturbance near Cooper is occurring.

The following conditions exist:

- 345 KV voltage is RISING.
- Main Generator exciter AMPs are RISING.
- NPPD System frequency is 59.8 hertz and steady.

What is causing the above indications?
What voltage regulator switch position is directed by Procedure 5.3GRID for these conditions?

A. The grid
OFF

B. The grid
ON

C. The voltage regulator
OFF

D. The voltage regulator
ON

Answer:

C. The voltage regulator
OFF

Explanation:
Actions in this procedure to place the voltage regulator to OFF are mandated under conditions indicative of a CNS voltage regulator oscillation causing 345 kV voltage variations. If the grid voltage is oscillating and voltage regulator is working as designed, as grid voltage lowers regulator will raise field amps in an attempt to maintain terminal voltage. If regulator is causing the oscillations, field amps rising will cause 345 kV volts to rise. So, if 345 kV voltage rises as field amps lower, regulator is working properly and there is no need to transfer to the base adjuster. If 345 kV voltage rises as our field amps rise, then our regulator is driving the voltage oscillations and transferring to the base adjuster should stabilize conditions.

Distracters:

A. This answer is incorrect because the voltage regulator is causing the problem. This answer is plausible because the voltage regulator switch position is correct. The candidate who does not realize rising voltage regulator output and corresponding grid voltage rising is caused by the voltage regulator would select this answer.

B. This answer is incorrect because the voltage regulator is causing the problem and 5.3GRID directs placing the voltage regulator switch in OFF. If the candidate does not understand that the current condition on the grid is occurring due to the CNS main generator voltage regulator failing high in automatic they may chose this answer.

D. This answer is incorrect because the voltage regulator is causing the problem so its control switch must be placed in OFF. If the candidate does not recall the correct procedure guidance from 5.3GRID would select this answer.

Technical Reference(s): 5.3GRID Degraded Grid Voltage, Rev. 41

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: INT032-01-31 CNS Abnormal Procedures (RO) Electrical

S. Given plant condition(s) and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:	Bank #	_______
	Modified Bank #	_______
	New	___X___
Question History:	Last NRC Exam	_______
Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	__X__
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295002 2.4.50
Importance Rating	4.2

295002 Loss of Main Condenser Vacuum:

2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Question: 21

A Plant is operating at full power in winter months. The following conditions exist:

- Alarm B-1/B-3 TG LOW VACUUM PRE TRIP sounds.
- Main Generator load is 805 MWe.
- MS-PI-72A, A VACUUM is indicating 23.5" Hg Vac. and slowly degrading.
- MS-PI-72B, B VACUUM is indicating 23.5" Hg Vac. and slowly degrading.

(1) Where is the alarm setpoint validated?
(2) What action is directed by B-1/B-3, when vacuum degrades to 23" Hg?

A. (1) CONDENSER PRESSURE TRIP GRAPH on DEH HMI.
(2) Scram and trip the turbine.

B. (1) CONDENSER PRESSURE TRIP GRAPH on RVLC HMI.
(2) Scram and trip the turbine.

C. (1) CONDENSER PRESSURE TRIP GRAPH on RVLC HMI.
(2) Perform Rapid Power reduction.

D. (1) CONDENSER PRESSURE TRIP GRAPH on DEH HMI.
(2) Perform Rapid Power reduction.

Answer:

A. (1) CONDENSER PRESSURE TRIP GRAPH on DEH HMI.
(2) Scram and trip the turbine.

Explanation:
The main condenser low vacuum alarms (Pre-trip and Trip) are received from the ALARMS program module of the Trip Tricon unit. The DEH HMI contains a dynamic display, CONDENSER PRESSURE TRIP GRAPH. The control room operator verifies the alarm by noting the current plant operating status. With the condition given, the CONDENSER PRESSURE TRIP GRAPH indicates the operating point very close to the 8.25 absolute

pressure trip point. Alarm B-1/B-3 directs tripping the main turbine if condenser vacuum cannot be maintained above 23" Hg.

From ARP 2.3 B-1 (Actions for B-1/B-3 TG LOW VACUUM PRE TRIP annunciator)

5. OPERATOR OBSERVATION AND ACTION

NOTE – Main turbine trips on main condenser low vacuum which is dynamically calculated based on absolute pressure and megawatts.

5.1 Check vacuum indication on Panel B to verify alarm is valid.

5.2 Monitor dynamic graph on CONDENSER PRESSURE TRIP GRAPH.

5.3 **IF vacuum cannot be maintained ≥ 23" Hg, THEN perform following:Ⓟ[1]**

5.3.1 **IF Annunciator 9-5-2/C-4 clear, THEN SCRAM and enter Procedure 2.1.5**.

5.3.2 Trip main turbine.

5.3.3 IF reactor was not scrammed, THEN enter Procedure 2.2.77.

5.4 IF alarm is valid, THEN take action per Procedure 2.4VAC.Ⓟ[1]

Distracters:

B. This answer is incorrect because the Main Turbine low vacuum pre trip alarm cannot be verified on the RVLCS HMI. This answer is plausible because the RVLCS HMI can display condenser vacuum and the RFP turbine receives a trip on loss of vacuum. The candidate who believes the vacuum pre-trip alarm can be verified on the RVLCS HMI would select this answer.

C. This answer is incorrect because rapid power reduction is not the correct action to take and the RVLCS HMI is not the correct place to verify the alarm. The abnormal procedure 2.4VAC, Loss of Condenser Vacuum provides direction to lower reactor power but does not direct rapid power reduction so this answer is plausible. The alarm procedure, B-1/B-3 does not direct lowering reactor power, but does provide direction to scram the reactor if vacuum is below 23" Hg. If the candidate cannot recall the scram action from B-1/B-3 would select this answer. This answer is plausible because rapidly lowering reactor power aids in condenser vacuum recovery.

D. This answer is incorrect because rapid power reduction is not the correct action to take. This answer is plausible because the correct HMI is listed and lowering reactor power aids in condenser vacuum recovery. The candidate who knows the correct location to verify the alarm and knows reactor power reduction helps in mitigating the low vacuum would select this answer.

Technical Reference(s): 2.4VAC LOSS OF CONDENSER VACUUM, Rev. 25
ALARM PROCEDURE 2.3_B-1 PANEL B - ANNUNCIATOR B-1, Rev. 34

Proposed references to be provided to applicants during examination: NONE

Learning Objective: INT032-01-32 CNS Abnormal Procedures (RO) Off Gas/Vacuum

J. Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

K. Given plant condition(s), determine from memory if a Main Turbine trip is required due to the event(s).

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295013 AK1.03
	Importance Rating	3.0

295013 High Suppression Pool Temp.:

Knowledge of the operational implications of the following concepts as they apply to high suppression pool temperature:
(CFR: 41.8 to 41.10)

AK1.03 Localized heating

Question: 22

The plant is operating at 100% power.

- All MSIVs close causing a Reactor Scram.
- The operator takes manual control of pressure using the SRVs.

Per Procedure 2.2.1, Nuclear Pressure Relief System, why are the SRVs alternated while controlling pressure?

Prevents…

A. bulk torus overheating.

B. localized torus overheating.

C. valve failure due to excessive cycling.

D. valve failure due to excessive valve temperature.

Answer:

B. localized torus overheating.

Explanation:

The SRV discharge is into the suppression pool. The energy is transported to a local area near the T-Quenchers and, if the same valve is re-opened repeatedly, the area in the pool does not have a chance of mixing and a local high temperature results.

From 2.2.1 Nuclear Pressure Relief System

6. PRECAUTIONS AND LIMITATIONS

6.1 When manually operating main steam relief valves, alternate use of valves as indicated on Panel 9-3. Avoid prolonged usage of individual valves (> 2 minutes) to minimize overheating in localized areas. Space opening of each safety/relief valve by ≥ 3 seconds.

Distracters:

A. This answer is incorrect because the bulk temperature is not as directly effected as the localized area. The candidate may pick this if he/she believes the bulk heating is the reason for re-opening the same SRV. This answer is plausible because the energy is added to the same body of water no matter which SRV is opened.

C. This answer is incorrect because the procedure requires a minimum of 3 seconds between re-opening the same SRV. If the SRV was re-opened without the 3 second wait, then the SRV tailpipe could sustain damage due to the water slug being drawn into the tailpipe because the vacuum breakers are not given enough time to clear the column of water. The candidate may pick this if he/she does not recall the wait period between valve re-opening.

D. This answer is incorrect because the valve is designed to withstand the temperature of the fluid flowing through it. The candidate may select this answer if he/she does not understand that using a single SRV would not result in valve overheating.

Technical Reference(s): SOP 2.2.1 Nuclear Pressure Relief System, Rev. 38

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR002-16-02 OPS Nuclear Pressure Relief

4. Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters:

   e. Suppression pool temperature

Question Source:	Bank #	_______
	Modified Bank #	_______
	New	___X___
Question History:	Last NRC Exam	_______
Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295002 AK2.02
	Importance Rating	3.1

295022 Loss of CRD Pumps:

Knowledge of the interrelations between loss of CRD pumps and the following:

AK2.02 CRD mechanism

Question: 23

A plant startup is in progress with pressure set at 926 psig and reactor power at 8%.

- CRD Pump A trips and cannot be restarted.
- All attempts by the crew to start CRD Pump B are unsuccessful.

What is the consequence to the Control Rod Drive Mechanism (CRDM) if operation continues with these conditions?

A. Reduced seal life.

B. Shorter scram times.

C. Cooling water orifice plugging.

D. Failure of the piston tube assembly.

Answer:
A. Reduced seal life.

Explanation:

From 2.4CRD

Loss of or inadequate cooling water to the CRDMs causes the inability to move rods and elevated CRDM temperatures. **The CRDMs can operate without cooling water flow but seal life may be shortened by exposure to reactor operating temperatures.** CRDM temperatures over 350°F may result in a measurable delay in scram response times. A rise to 400°F could result in up to a 0.150 second rise in the 90% insertion time for an otherwise normally performing CRD. The evaluation of Scram time Tau correction is performed by Procedure 10.35 when temperature exceeds 350°F.

Distracters:

B. This answer is incorrect because scram times may be longer when drive temperatures are above 350°F. The candidate may confuse elevated drive temperatures with scram times being longer. This answer is plausible because scram times are effected by elevated drive temperatures due to loss of CRD cooling water.

C. This answer is incorrect because cooling water plugging is not an immediate issue. This answer is plausible because suddenly stopping cooling water flow could cause impurities settled in the mechanism to become disturbed. The screen at the mechanism cooling water inlet should filter out the larger debris that could cause orifice plugging which makes this answer plausible.

D. This answer is incorrect because there is no concern with the piston tube assembly. Elevated drive temperatures result in elevated seal temperatures and the candidate may believe seal failure could result in elevated d/p across the piston tube assembly and damage to the piston tube. This answer is plausible because piston tube assembly damage can occur in CRDMs.

Technical Reference(s): 2.4CRD CRD Trouble, Rev 15

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: COR002-04-02 Control Rod Drive Hydraulics

12. Given a specific CRDH system malfunction, determine the effect on any of the following:

c. Control Rod Drive Mechanisms (CRDMs)

Question Source: Bank # 19962
Modified Bank #
New

Question History: Last NRC Exam _______

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295033 EK3.02
Importance Rating	3.5

295033 High Secondary Containment Area Radiation Levels:

Knowledge of the reasons for the following responses as they apply to high secondary containment area radiation levels:

EK3.02 Reactor SCRAM

Question: 24

The plant is operating at 100% power when an un-isolable RCIC steam line failure results in high radiation in the secondary containment.

What is the reason for inserting a reactor scram before secondary containment radiation levels reach the Max Safe value?

A. Prevents any component environmental qualification (EQ) from being exceeded.

B. Ensures an emergency depressurization of the Reactor Pressure Vessel is NOT required.

C. Precludes exceeding secondary containment positive pressure design and actuation of the building 1001' blow-out panels.

D. Reduces the driving head of a primary system discharge into secondary containment in anticipation of an emergency depressurization.

Answer:

D. Reduces the driving head of a primary system discharge into secondary containment in anticipation of an emergency depressurization.

Explanation:

With a primary system discharging into secondary containment, the EPGs give the basis for when to scram the plant and conduct an emergency depressurization. Scramming reduces the driving head of a primary system that is discharging into the secondary containment and in anticipation of performing an emergency depressurization if radiation levels continue to rise.

From EPG Rev 3:

Primary systems comprise the pipes, valves, and other equipment physically connected to the RPV such that a reduction in RPV pressure will effect a decrease in the flow of steam or water being discharged through an unisolated break in the system.

A reactor scram is initiated through entry of the RPV Control guideline to reduce the primary system discharge into secondary containment and in anticipation of possible RPV depressurization in Step SC/R-2.2. If a discharge from a primary system is the source of radioactivity, the action that is directed in Step SC/R-2.1 should be adequate to terminate any further increase in secondary containment radiation levels.

Distracters:

A. This answer is incorrect because the scram is not based on protecting EQ equipment in secondary containment. Secondary containment has numerous EQ equipment that are designed for to withstand DBA, and Special Events so the candidate may believe these actions are need to protect EQ equipment. This answer is plausible because protecting safe shutdown equipment is a prudent action to take.

B. This answer is incorrect because the scram will not always preclude emergency depressurization. A scram is entered to lessen the amount of primary system discharging into secondary containment in an attempt to prevent exceeding MSO levels of a given parameter. The candidate who knows scramming will lessen the primary system discharging into secondary containment may preclude emergency depressurization would select this answer. This answer is plausible because the action is taken to attempt to preclude emergency depressurization.

C. This answer is incorrect because the reactor building blow-out panel actuation is in response to environmental conditions. This answer is plausible because there are blow-out panels in the steam tunnel area that are designed to open on a HELB event. However, the blow-out panels on the refueling floor area are designed for tornado events. The candidate who does not fully understand the design of the blow-out panels would select this answer.

Technical Reference(s): EOP-5A Secondary Containment Control, Rev. 15

Proposed references to be provided to applicants during examination: ___NONE_______

Learning Objective: INT008-06-17 EOP Flowchart 5A Secondary Containment and Radioactivity Release Control

7. Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Question Source: Bank # ______
Modified Bank # ______
New __X____

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	_
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295034 EA1.03	
	Importance Rating	4.0	

295034 Secondary Containment Ventilation High Radiation / 9 – Ability to operate and / or monitor the following as they apply to secondary containment ventilation high radiation: (CFR: 41.7) EA1.03 Secondary containment ventilation

Question: 25

Following a RWCU demineralizer vessel rupture and subsequent resin spill, the Reactor Building Vent Exhaust Radiation Monitoring system channels are reading as follows:

- Channel A 11 mr/hr
- Channel B 6 mr/hr
- Channel C 5 mr/hr
- Channel D 10 mr/hr

There are no plant system responses to this condition.

What are the required control room operator responses for secondary containment ventilation systems?

A. Start both SGT trains only.

B. Secure RB ventilation only.

C. Start one SGT train; secure RB ventilation.

D. Start both SGT trains, secure RB ventilation.

Answer:

D. Start both SGT trains, secure RB ventilation.

Explanation:

Radiation Monitors A and C are in division 1 while B and D are in division 2. Either division will trip upon receipt of one (1) Radiation monitor upscale trip at 10 mr/hr. With a trip signal present in both divisions, a full Group 6 isolation will be initiated.

The Group 6 signal isolates Reactor BLDG Ventilation which causes the initiation of both trains of Standby Gas Treatment.

Distracters:

A. This is incorrect because reactor building ventilation should have secured so the operator must perform that task. This option is plausible because there are instances where starting both trains of SGT is the action to take for ventilation failures (e.g., loss of stack dilution fans).

B. This answer is incorrect because the action to start both SGT trains is also required. This answer is plausible because RB ventilation should have secured itself.

C. This answer is incorrect because the action is to start both SGT trains. This answer is plausible because this is the system lineup that would be present after the secondary containment pressure is verified and one train of SGT is secured.

Technical Reference(s): SOP 2.2.47, HVAC Reactor Building, Rev. 50
IOP 4.7.5, Reactor Building Vent Exhaust Radiation Monitoring System, Rev 18

Proposed references to be provided to applicants during examination: ___None_________

Learning Objective:

COR001-08-01

6. Describe the interrelationships between HVAC systems and the following:
   e. Process Radiation Monitoring system

11. Describe the HVAC design features and interlocks that provide for the following:
   b. Secondary containment isolation

COR001-18-01

5. Describe the interrelationship between the RM system and the following:
   r. Reactor Building Ventilation system

8. Describe the Radiation Monitoring system design feature(s) and/or interlock(s) that provide for the following:
   b. Automatic action to contain the radioactive release in the event that the predetermined release rates are exceeded.

Question Source: Bank # _______
Modified Bank # _6021__ (See attached)
New _______

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge __X__

10 CFR Part 55 Content: 55.41 _(7)_

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295035 EA2.02	
	Importance Rating	2.8	

295035 Secondary Containment High Differential Pressure /5 – Ability to determine and/or interpret the following as they apply to secondary containment high differential pressure: (CFR: 41.8) EA2.02 Off-site release rate: Plant-Specific

Question: 26

The plant is conducting refuel operations involving the movement of recently irradiated fuel assemblies in the secondary containment.

- A fuel bundle is dropped on the top of the core.
- A Group 6 isolation occurs due bundle damage.
- Calm winds are indicated on the SPDS Weather Display.
- HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET fail to fully isolate.
- Reactor Building Average dp indicates -0.04 in. wg and stable.

Which of the following identifies the actual Off-Site radiological release rate response?

A. ERP release rate rises.

B. ERP release rate lowers.

C. Reactor Building monitored release rate rises.

D. Reactor Building unmonitored release rate rises.

Answer:

A. ERP release rate rises.

Explanation:

A fuel handling accident involving handling of recently irradiated fuel inside of the secondary containment is one of the two principal accident scenarios for which credit is taken for secondary containment operability. Typically the Secondary Containment requires ≥0.25 inches of vacuum water gauge to maintain OPERABILITY. In this case however, the Reactor Building Average dp is below 0 inches water gauge (negative) and stable so both Standby Gas Trains are able to maintain negative building pressure under calm wind conditions and no RB unmonitored release is indicated. Since the Group 6 isolation occurred, the RB exhaust fans have terminated the release from the reactor building exhaust plenum. The ERP release rate rises because both SGTs are operating and providing an increased flow due to the failure of the

HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET to fully isolate and the ventilation flow that was being processed through the RB exhaust plenum is now routed to the ERP through the SGT system.

Distracters:

B. This answer is incorrect because the ERP release rises based on the reroute of radioactive gases previously being released through the RB ventilation plenum. This choice is plausible because the student may think that the charcoal filters are adequate treatment for the release from the fuel bundle, but they have minimal effect on the noble gases being released. It should be noted that the RB ventilation response time is designed to isolate before the radioactive gases actually exit the RB exhaust stack.

C. This answer is incorrect because the RB ventilation system has isolated. This choice is plausible if the candidate thinks that the failure of the HV-MO-262 MG SET-1A INLET and HV-AO-263 MG SET-1A INLET to fully isolate contribute to a rise in Reactor Building ventilation release rate. In this case, the building is being maintained negative by the SGT system and not the RB ventilation system.

D. This answer is incorrect in this instance because the SGT system in conjunction with the partial integrity of the Secondary Containment is adequate to maintain a negative RB pressure. This answer is plausible if the candidate does not recognize that the RB pressure is sufficiently negative to prevent an unmonitored release for the current wind conditions.

Technical Reference(s): SOP 2.2.47, HVAC Reactor Building, Rev 50
(Attach if not previously provided) SOP2.2.73, Standby Gas Treatment System, Rev 52.

Proposed references to be provided to applicants during examination: _None___________

Learning Objective:

COR001-08-01
13. Briefly describe the following concepts as they apply to HVAC:
    a. Airborne contamination control

COR002-28-02R22
1. State the purpose of the following items related to the Standby Gas Treatment (SGT) System:
    e. High efficiency inlet filter (HEPA)
    f. Activated carbon iodine adsorber (charcoal filter)
    e. High efficiency final filter

7. Given a specific Standby Gas Treatment System malfunction, determine the effect on any of the following:
    a. Secondary Containment differential pressure
    b. Off-Site release rate

COR002-03-02R30
7. Describe the interrelationship between Secondary Containment and the following:
    a. Reactor Building Ventilation
    c. SGT
    d. ERP

19. Predict the consequences of the following items on Secondary Containment:
    c. High airborne radiation
25. Predict the consequences of a malfunction of the following on Secondary Containment:
    a. Reactor Building Ventilation

Question Source:	Bank #	________
	Modified Bank #	________
	New	___X___
Question History:	Last NRC Exam	____________
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 _(9)_	

Comments:

LOD 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	500000 2.4.21	
	Importance Rating	4.0	

500000 High CTMT Hydrogen Conc. / 5 – 2.4.21 Knowledge of 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, radioactive release control, etc. (CFR: 41.7)

Question: 27

Which of the following identifies the Drywell Hydrogen concentration that requires venting and purging Primary Containment (PC) AND whether ODAM Release Rates are allowed to be exceeded IAW Procedure 5.8.21 {PC Venting AND Hydrogen Control (Less Than Combustible Limits)}?

PC H_2 concentration at...

A. 0.34% requires venting and purging the Drywell within ODAM limits.

B. 0.34% requires venting and purging the Drywell, exceeding ODAM limits is allowed.

C. 1% requires venting and purging the Drywell within ODAM limits.

D. 1% requires venting and purging the Drywell, exceeding ODAM limits is allowed.

Answer:

C. 1% requires venting and purging the Drywell within ODAM limits.

Explanation:
Requires knowledge of containment H_2 concentration which requires venting (Rad Release Control) and the impact on offsite release. EOP 3A (PCCP) requires venting & purging PC when H_2 concentration reaches 1% only if offsite radioactivity release rate is expected to remain below the offsite release rate limits specified in ODAM.

Distracters:

A. This answer is incorrect due to H_2 concentration being less than 1%. This choice is plausible if the H_2 Hi & Hi Hi alarm setpoints are confused (34% is 10% of the Hi Hi alarm setpoint). The candidate who confuses Hi & Hi Hi H_2 alarm setpoints and correctly recognizes release within ODAM limits would select this option.

B. This answer is incorrect due to H_2 concentration being less than 1% and having release above ODAM limits. This choice is plausible if the H_2 Hi & Hi Hi alarm setpoints are confused (34% is 10% of the Hi Hi alarm setpoint) and if venting PC is confused with the emergency release rate which requires a General Emergency (above ODAM limit). The candidate who confuses Hi & Hi Hi H_2 alarm setpoints and confuses emergency release above ODAM limits would select this option.

D. This answer is incorrect due to having releases above ODAM limits. This choice is plausible if venting PC is confused with the emergency release rate which requires a General Emergency (above ODAM limit). The candidate who confuses Hi & Hi Hi H_2 alarm setpoints and confuses emergency release above ODAM limits would select this option.

Technical Reference(s):
EOP 5.8.21 {PC Venting AND Hydrogen Control (Less Than Combustible Limits)}, Rev 18
EOP-3A (Primary Containment Control), Rev 15
5.9 H2O2 { Primary Containment Combustible Gas Control (SAG 3)}, Rev 8
Procedure 5.8.22

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-03-02

12. Describe the Containment design features and/or interlocks that provide for the following:
    d. Hydrogen control

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
10 CFR Part 55 Content:	55.41	(7)

Comments:

DIF 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A3.03	
	Importance Rating	3.7	

203000 RHR/LPCI: Injection Mode – Ability to monitor automatic operation of the RHR/LPCI: injection mode (plant specific) including: (CFR: 41.7) A3.03 Pump discharge pressure

Question: 28

With the plant operating at rated power a LOCA occurs.

The following conditions exist:

- Reactor pressure is 600 psig and lowering.
- Drywell pressure is 5.2 psig and rising slowly.
- Torus pressure is 4.0 psig and rising slowly.

When does **RHR(LPCI) injection flow** into the RPV **first** occur?

A. 550 psig

B. 435 psig

C. 265 psig

D. 105 psig

Answer:

C. 265 psig

Explanation:

With drywell pressure greater than 1.84 psig a LPCI initiation signal is present. The RHR pumps receive a start signal and operate on minimum flow. As reactor pressure continues to lower the RHR inboard injection valves open when reactor pressure reaches 436 psig. Flow however does not occur until reactor pressure falls below the shutoff head of the RHR pumps at 265 psig. At or below this pressure indications of flow from the RHR would first occur.

Distracters:

A. This option is incorrect because, with reactor pressure at 550 psig, the RHR pumps are not injecting, as this pressure is greater than the shutoff head of the RHR pumps. As reactor pressure lowers to this value if the condensate and condensate booster pumps are operating there would be flow from this source. A candidate who has seen these conditions during training in the in the simulator may confuse the where the source of injection came from and choose this option.

B. This option is incorrect because, with reactor pressure at 436 psig, the RHR pumps are not injecting, as this pressure is greater than the shutoff head of the RHR pumps. The RHR Inboard injection valves however are interlocked to open at this pressure establishing a flow path from the RHR pumps to the reactor vessel. A candidate may choose this answer however believing that flow begins when the inboard injection valve opens.

D. This option is incorrect because, with reactor pressure at 265 psig, the RHR pumps are injecting. This is the pressure where pressure main condensate pumps inject. The candidate may recall the condensate pump injection pressure and confuse this with the point where RHR injects..

Technical Reference(s): USAR Section VI, Table VI-5-4. Plant ECCS Parameters
(Attach if not previously provided) ____________
(including version/revision number) ____________

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-23-02, OPS Residual Heat Removal System

4. Describe the interrelationship between the RHR system and the following:

   n. Reactor pressure

Question Source:	Bank #	______
	Modified Bank #	______
	New	X
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 A.4.07	
	Importance Rating	3.7	

205000Shutdown Cooling – Ability to manually operate and/or monitor in the control room: (CFR: 41.7) A4.07 Reactor temperatures (moderator, vessel, flange)

Question: 29

Residual Heat Removal (RHR) loop A is aligned for shutdown cooling mode of operation with RHR pump A operating. The following conditions exist:

- The vessel head is tensioned.
- B Reactor Recirculation pump is operating.
- SDC system flow is 5000 gpm.
- Temperatures are logged as follows:

TIME	RHR-TR-131, CH 9 (RHR HX inlet temp	NBI-TR-89, CH 9 (reactor vessel metal temp)	NBI-TR-89, CH6 (reactor vessel flange temp)
0100	210ºF	218ºF	218ºF
0115	188ºF	192ºF	195ºF
0130	166ºF	170ºF	173ºF
0145	143ºF	145ºF	153ºF

What RHR action is appropriate and why?

A. Throttle CLOSED RHR-MO-27A, Inboard Injection Valve, to reduce the cooldown rate.

B. Throttle OPEN RHR-MO-66, RHR Heat Exchanger Bypass Valve, to reduce the cooldown rate.

C. Throttle CLOSED RHR-MO-27A, Inboard Injection Valve, to ensure accurate temperature indication at the RHR HX Inlet.

D. Throttle OPEN RHR-MO-66, RHR Heat Exchanger Bypass Valve, to ensure accurate temperature indication at the RHR HX Inlet.

Answer:

B. Throttle OPEN RHR-MO-66, RHR Heat Exchanger Bypass Valve, to reduce the cooldown rate.

Explanation:

When the RHR system is placed in operation for SDC the heat up and cooldown rates are adjusted to average heatup/cooldown rate ≤ 90°F/hr averaged over any 1 hour period. RHR-MO-27A, RHR-MO-66A and RHR-MO-12A are the valves that are procedurally adjusted to manipulate cooldown rate. From the data provided, if the cooldown rate is continued at the current rate, the 90°F/hr administrative limit for cooldown will be exceeded. The cooldown rate for the reactor vessel metal temperature is currently excessive and if the current rate continues the cooldown rate will exceed the limit.

Opening RHR-MO-66A will reduce the cooldown rate. Although both RHR-MO-12 and 27 could be closed to reduce cooldown the SDC flow is already low and cannot be lowered to less than 5000 gpm and closing either of these valves would reduce SDC flow.

A. This option is incorrect because even though the cooldown rate is excessive, the conditions given have SDC flow at 5000 gpm and so throttling closed either RHR-MO-27A or 12A would reduce flow rate and would therefore be inappropriate with flow at the low limit. The candidate who fails to completely evaluate all the provided conditions may choose this option because this is a procedural method for reducing the cooldown rate.

C. This option is incorrect because the cooldown rate is excessive. The temperatures are trending together correctly so the change needed is to reduce the cooldown rate. Surveillance Procedure 6.RCS.601 requires monitoring of various temperatures during the cooldown to ensure proper trending so it a candidate may believe that the trends are not correct and that a SDC manipulation is required for accurate indication. MO-27 is a valve that can be manipulated to adjust SDC flow. However since flow is already at 5000 gpm this action is inappropriate because it would cause flow to go below 5000 gpm.

D. This option is incorrect because the cooldown rate is excessive which requires action. Surveillance Procedure 6.RCS.601 requires monitoring of various temperatures during the cooldown and if the temperatures are not trending together then it requires the manipulation of MO-66 to ensure that the water temperature measured is accurate. Because this is a possible manipulation that may be required, a candidate may believe that these values are not trending correctly and choose this answer.

Technical Reference(s):
SP 6.RCS.601, RCS Heatup/Cooldown Rate Monitoring, Rev 21
SOP 2.2.69.2, RHR System Shutdown Operations, Rev 89

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-23-02

6. Given an RHR control manipulation, predict and explain changes in the following:
   a. Heat exchanger temperature and flow
   d. Reactor parameters (level, pressure, temperature)

Question Source:	Bank #	________
	Modified Bank #	________
	New	___X___
Question History:	Last NRC Exam	_____________
Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	___X___
10 CFR Part 55 Content:	55.41 _(7)_	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	206000 2.2.39	
Importance Rating	3.9	

206000 HPCI – 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7)

Question: 30

The plant is operating at 10% of rated power.

HPCI-MO-15, STM SUPP INBD ISOL VLV is found closed and will not open with its control switch.

What is the required action?

A. Enter Technical Specification 3.0.3 immediately.

B. Verify all ADS SRVs are Operable within 1 hour.

C. Verify RCIC system is Operable by administrative means within 1 hour.

D. Isolate HPCI by deactivating HPCI-MO-15 and HPCI-MO-16 within 1 hour.

Answer:
C. Verify RCIC system is Operable by administrative means within 1 hour.

Explanation:

Per TS 3.5.1, CONDITION C; If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI System must be restored to OPERABLE status within 14 days. In this condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low-pressure ECCS injection/spray subsystems in conjunction with ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY within 1 hour is therefore required when HPCI is inoperable. RCIC is required to be determined to be operable by administrative means within one hour.

Distracters:

A. This option is incorrect because entry into 3.0.3 is not required because only one ECCS system is inoperable. If two were inoperable then this option would be correct. The candidate may remember that there is an immediate entry into 3.0.3 required for an inoperable ECCS. Since it is a less than 1-hour specification associated with HPCI a candidate may choose this option.

B. This option is incorrect because verifying ADS operability is not required by this specification. Since functionally ADS provides a backup to the HPCI system, (in conjunction with low pressure ECCS) and because RCIC is not an ECCS system, a candidate may believe that verifying that ADS is operable is the required action and would therefore choose this option. This option is a common misconception.

D. This option is incorrect because this action is not required. The actions for a primary containment isolation valve inoperable are contained in 3.6.1.3. But since only one valve is inoperable and it is closed this specification does not apply. However this would be the action required if both HPCI isolation valves had failed which is why a candidate may choose this answer.

Technical Reference(s): Technical Specification 3.5.1, ECCS-Operating
(Attach if not previously provided) SOP 2.2.33, High Pressure Coolant Injection System, Rev 77

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT007-05-06, OPS Tech Specs 3.5, Emergency Core Cooling systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System

3. Given a set of plant conditions that constitutes non-compliance with a Section 3.5 LCO, determine the Actions that are required.

4. From memory, in MODES 1, 2, and 3, state the actions required in ≤ one hour if HPCI System is inoperable **or** two or more low pressure ECCS injection/spray subsystems inoperable **or** HPCI System and one or more ADS valves are inoperable (LCO 3.5.1).

Question Source: Bank # ______
Modified Bank # ______
New ___X___

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge _X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _10_

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	209001 K1.01	
Importance Rating	3.1	

209001 LPCS – Knowledge of the physical connection and/or cause-effect relationships between low pressure core spray and the following: (CFR: 41.2 – 41.9) K1.01 Condensate storage tank: Plant-Specific

Question: 31

What is the alternate suction source for Core Spray pump A, and why is it aligned to this source?

A. ECST, to raise torus level in Modes 1/2/3.

B. CST A, to raise torus level in Modes 1/2/3.

C. ECST, for core reflood capability when the torus is drained in Modes 4/5.

D. CST A, for core reflood capability when the torus is drained in Modes 4/5.

Answer:

D. CST A, for core reflood capability when the torus is drained in Modes 4/5.

Explanation:
Per USAR Chapter VI Section 4 the CST provides an alternate suction source to CS. The suction to the CS pumps can also be lined up to Condensate Storage Tank (CST) 1A. CNS Technical Specifications allow refueling operations to be conducted with the suppression pool drained provided an operable CS or LPCI subsystem is aligned to take a suction on CST 1A, containing at least 150,000 gallons. In this condition, the reactor vessel is depressurized and the CS subsystem provides a core reflooding capability.

Distracters:

A. This option is incorrect because CS A is not capable of being aligned to the ECST. The other core cooling systems (HPCI and RCIC) can be aligned to the ECST, which is why a candidate may believe that CS is capable of this alignment. Additionally, the purpose of the alternate source for the CS is to provide for core reflood when in modes 4 and 5. But under certain conditions, the Core Spray system can be used to fill the torus (with pressure maintenance) so a candidate may choose this option.

B. This option is incorrect because the purpose of the alternate suction source for CS A is not to provide the capability to fill the torus it is to allow reflood in modes 4 and 5 when the torus is not available. Since the torus can be filled with the CS system (from pressure maintenance not the alternate suction path) a candidate may believe that this is the purpose.

C. This option is incorrect because the alternate suction source is from the CST not the ECST. But other core cooling systems (HPCI and RCIC) can be aligned to the ECST, which is why a candidate may believe that CS is capable of this alignment.

Technical Reference(s): SOP 2.2.9, Core Spray System, Rev 76
(Attach if not previously provided) COR002-06-02, Core Spray System, Rev 87
(including version/revision number) USAR Chapter VI Section 4

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-06-02, Core Spray System
3. Describe the interrelationships between the Core Spray and the following:

a. Condensate Storage Tank

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 8	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	259002 K3.02	
Importance Rating	3.7	

259002 Reactor Water Level Control - Knowledge of the effect that a loss or malfunction of the reactor water level control system will have on following: (CFR: 41.7 / 45.4) (CFR: 41.7 / 45.5 to 45.8) - K3.02 Reactor feedwater system

Question: 32

The plant is operating at near rated power when a loss of both RVLC/RFPT CORE switches causes RFPT control to transfer to MDEM.

How are the RFPs affected?

A. Speed lowers to idle speed.

B. Speed rises to upper automatic clamp.

C. Speed is held constant at current speed.

D. Speed lowers to minimum governor speed.

Answer:

C. Speed is held constant at current speed.

Explanation:

The loss of RVLC/RFPT CORE switches, which are part of the Reactor Vessel Level Control System, result in the RFPs transferring to MDEM. With the loss of the core switches, there is no control available from HMIs. Since the RFPs transfer to MDEM the RFP speed is held constant and no longer modulates to control level.

Distracters:

A. This option is incorrect because, with the failure of the core switches the controllers transfer to MDEM and now instead of RFP speed modulating it now locks at its current speed. A candidate could believe that with the loss of the switches that the output would be low as is the case with many analog controllers and that speed would therefore lower to idle speed (1000 RPM). As idle speed is an operationally significant speed during a RFP start a candidate may choose this option.

B. This option is incorrect because, when the RFPs transfer to MDEM, they do so at their current speed. But a candidate may believe that with the loss of the core switches that speed would rise to the upper clamp at 5800 RPM. Some control systems that suffer loss of input do go to their maximum values so a candidate who does not understand this system may choose this option.

D. This option is incorrect because, with the failure of the core switches the controllers transfer to MDEM and now instead of RFP speed modulating it now locks at its current speed. A candidate could believe that with the loss of the switches that the output would be low as is the case with many analog controllers and that speed would therefore lower to the minimum clamp on the governor (2000 RPM). This is a different speed than the idle in option "A" so the candidate who believes that this functions as do many analog controllers may choose this options because this minimum governor speed is an operationally significant value.

Technical Reference(s): Instrument Procedure 4.4.1, Reactor Vessel Level Control, Rev 7
COR002-32-02 Reactor Vessel Level Control, Rev 14

Proposed references to be provided to applicants during examination: __None__________

Learning Objective:
COR002-32-02 Reactor Vessel Level Control
9. Given a specific RVLC system malfunction, determine the effect on any of the following:

c. Feedwater System

Question Source: Bank # _______
Modified Bank # _______
New __X___

Question History: Last NRC Exam __________

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ___

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD: 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K2.02	
	Importance Rating	3.1	

211000 SLC – Knowledge of electrical power supplies to the following: (CR: 41.7) K2.02 Explosive valves

Question: 33

What is the power supply that is used to fire the "A" Standby Liquid Control (SLC) squib valve?

A. MCC K

B. MCC S

C. MCC M

D. 120 VAC CPP

Answer:
A. MCC K

Explanation:

MCC K is the power supply to the A SLC pump and the squib valve receives its power from the pump supply breaker.

Distracters:

B. This option is incorrect as this is the power supply to the B SLC pump and the B squib valve. A candidate may confuse which power supply is associated with which pump and may therefore choose this option. This answer is plausible because this power supply does power a SLC squib valve.

C. This option is incorrect because this is the listed power supply for SLC heat tracing. A candidate who knows SLC loads are powered from MCC M but is not certain of the squib valve power supply may choose this answer because of its association with the SLC system. This answer is plausible because this power supply does power components in the SLC system.

D. This option is incorrect because this is the power supply for the squib valve ready lights. A candidate who does not know the squib valves are fired by an auxiliary contact in the pump breaker may believe this circuit also powers the squib valves since it does provide power to a squib related component (squib ready lights). Note: that even though this power supply appears different than the other options it is highly plausible because it powers actual squib components just not to power to fire the squib.

Technical Reference(s):

1. Ops Standby Liquid Control/COR0022902 Rev. 20

2. Procedure 2.2.23 Rev. 53, 120/240 VAC Instrument Power Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022902 R20
13. State the electrical power supply to the following SLC components:
    b. Squib valves

Question Source: Bank # ____________
Modified Bank # 32 on 2014 NRC Exam (See attached)
New

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 41.7

Difficulty: 2

211000 SLC

K.2 Knowledge of electrical power supplies to the following:

K2.02: Explosive Valves

Question 32

What is the power supply that is used to fire the “B” Standby Liquid Control (SLC) squib valve?

A. MCC K

B. MCC S

C. MCC M

D. 120 VAC CPP

Correct Answer:

B. MCC S

Explanation (Why distractors are incorrect and why correct answer is correct):

MCC S is the power supply to the B SLC pump and the squib valve receives its power from the pump supply breaker.

Each of the possible answers is a listed power supply for SLC components and is therefore plausible if the candidate is uncertain of the correct answer.

a. This option is incorrect because this is the power supply for SLC squib valve A and may be selected if the candidate is confused as to which source supplies which valve.
b. Correct. This is the listed power supply for SLC squib valve B from the technical references.
c. This option is incorrect because this is the listed power supply for SLC heat tracing. A candidate that knows SLC loads are powered from MCC M but is not certain of the squib valve power supply may choose this answer because of its association with the SLC system.
d. This option is incorrect because this is the power supply for the squib valve ready lights. A candidate that does not know the squib valves are fired by an auxiliary contact in the pump breaker may believe this circuit also powers the squib valves since it does provide power to a squib related component (squib ready lights). **Note: that even though this power supply appears different than the other options it is highly plausible because it powers actual squib components just not to power to fire the squib.**

Technical Reference(s):

1. Ops Standby Liquid Control/COR0022902 R20

2. Procedure 2.2.23 R 53 120/240 VAC Instrument Power Distribution System

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022902 R20
13. State the electrical power supply to the following SLC components:

b. Squib valves

Question Source: Bank # ________
Modified Bank # ______________ (Note changes or attach parent)
New _X

Question Cognitive Level: Memory or Fundamental Knowledge _X_____
Comprehension or Analysis ________

10 CFR Part 55 Content: 41.7

Difficulty: 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K4.02	
	Importance Rating	3.5	

212000 RPS – Knowledge of the reactor protection system design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.02 The prevention of a reactor SCRAM following a single component failure

Question: 34

The reactor is operating at 100% rated power with RPS power supplied by the RPS MG sets when the Reactor Protection System (RPS) MG Set A motor supply fused disconnect fuse blows.

(1) What is the status of the RPS A Electrical Protection Assemblies (EPA)?
(2) What is the status of RPS?

A. (1) Two (2) RPS A EPA Breakers opened.
   (2) Full reactor SCRAM.

B. (1) Two (2) RPS A EPA Breakers opened.
   (2) ½ scram on RPSPP1A.

C. (1) Four (4) RPS A EPA Breakers opened.
   (2) Full reactor SCRAM.

D. (1) Four (4) RPS A EPA Breakers opened.
   (2) ½ scram on RPSPP1A.

Answer:

B. (1) Two (2) RPS A EPA Breakers opened.
   (2) ½ scram on RPSPP1A.

Explanation:

RPS MG Set A generator supplies RPSPP1A. RPS MG Set B supplies RPSPP1B. The MG sets supply power to their respective RPS power panel via a pair of EPA in series. The EPAs ensure a pure and consistent power source for the sensitive RPS instrumentation which are energized and fail safe in the de-energized state. There is no automatic power transfer in the RPS system. Loss of power to the RPS MG Set causes two EPAs to trip on under-voltage which causes a half scram on the A side. Since MG set B remains energized, a full scram does NOT occur.

There are four EPA breakers for each RPS power supply. TWO series breakers for the MG Set and two series breakers for the alternate power supply. Only the two breakers associated with the degraded power supply will trip allowing power to be manually transferred without operation of any EPA Breakers.

Distracters:

A. This option is incorrect because RPSPP1B remains energized. The candidate could choose this option if he/she did not correctly identify that only one side of RPS was impacted. This answer is plausible because there are some single components that cause a full scram vice a divisional partial scram and the number of EPAs that trip is correct.

C. This option is incorrect because only the two breakers associated with the MG Set will trip because RPSPP1B remains energized. The candidate may choose this option if he/she did not identify that only two EPAs were impacted and did not correctly identify that only one side of RPS was impacted. This answer is plausible because there are some single components that cause a full scram vice a divisional partial scram.

D. This option is incorrect because only the two breakers associated with the MG Set will trip. The candidate may choose this option if he/she did not identify that only two EPAs were impacted and did not correctly identify that only one side of RPS was impacted. This answer is plausible because the partial scram is correct.

Technical Reference(s):	OPS Reactor Protection System COR002-21-02, Rev 23
(Attach if not previously provided)	SOP-2.2.22, Vital Instrument Power Supply, Rev 70
(including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-21-02, OPS Reactor Protection System

4. Describe the RPS design features and/or interlocks that provide for the following:

    b. Scram prevention following single component failure
    l. Under/over voltage and frequency protection

8. Given a specific RPS malfunction, determine the effect on any of the following:

    f. RPS logic channels

11. State the electrical power supplies to the following RPS components:

    a. RPS MG set
    b. RPS alternate power

Question Source:	Bank #		
	Modified Bank #	5210	(See attached)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__

10 CFR Part 55 Content: 55.41 _(7)_

Comments:

LOD 2

Original Bank Question:

QUESTION: 5210

The Reactor Protection System (RPS) MG Set A motor breaker has tripped causing RPSPP1A to be de-energized.

Which choice below describes the Electrical Protection Assembly (EPA) status **AND** actions required to restore power to RPSPP1A from the Alternate source for these conditions?

a. Two (2) RPS A EPA Breakers opened.
BOTH breakers must be re-closed prior to restoring power to RPSPP1A.

b. Two (2) RPS A EPA Breakers opened.
NEITHER breaker must be re-closed prior to restoring power to RPSPP1A.

c. Four (4) RPS A EPA Breakers opened.
ALL four (4) breakers must be re-closed prior to restoring power to RPSPP1A.

d. One (1) RPS A EPA Breaker opened.
The breaker does **NOT** need to be re-closed prior to restoring power to RPSPP1A.

ANSWER: 5210

b. Two (2) RPS A EPA Breakers opened.
NEITHER breaker must be re-closed prior to restoring power to RPSPP1A.

EXPLANATION OF ANSWER: b. Correct. There are four EPA breakers for each RPS power supply. TWO series breakers for the MG Set and two series breakers for the alternate power supply. Only the two breakers associated with the degraded power supply will trip allowing power to be transferred without operation of any EPA Breakers. a. Neither must be re-closed. c. Only the two associated with the MG Set will trip. d. Two breakers associated with the MG Set will trip.

REFERENCES: COR002-21-02, page 27-28, section IV.B, rev. 11. PR 4.5, page 8, section 4, rev. 21.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215003 K5.01	
Importance Rating	2.6	

215003 IRM – Knowledge of the operational implications of the following concepts as they apply to intermediate range monitor (IRM) system: (CFR: 41.5) K5.01 Detector operation

Question: 35

IRM "A" detector is installed with twice the argon fill pressure of the other IRM detectors.

How does this affect the operation of the "A" IRM versus the others when subjected to the same neutron field?

IRM "A" high high trip is…

A. less conservative and the downscale rod block is less conservative.

B. less conservative and the downscale rod block is more conservative.

C. more conservative and the downscale rod block is less conservative.

D. more conservative and the downscale rod block is more conservative.

Answer:

C. more conservative and the downscale rod block is less conservative.

Explanation:
The IRM detectors operate in the ionization region and their output (in a constant neutron flux) is effected primarily by the Argon pressure. The IRMs are not sensitive to small voltage changes because of their operation in the ionization region. Fission events in the detector cause ionizations as the fission fragments ionize the detector fill gas. If there is more argon in the detector then there will be more ions created by a given fission event so for IRM A each fission event causes a greater detector output. This means that the high high trip for that IRM occurs at a lower neutron flux (conservative) and because the IRM output is higher it also means that the downscale trip does not occur until a lower (non-conservative) neutron flux is reached.

A. This option is incorrect because the high high trip is more conservative because the output will reach the high high trip at a lower neutron flux. A candidate who believes that the additional argon in the detector would shield the ions from the anode (cathode), and who also believes that any reduction in detector output causes all associated trips to be less conservative.

B. This option is incorrect because the high high trip is more conservative because the output will reach the high high trip at a lower neutron flux. A candidate who believes that the additional argon in the detector would shield the ions from the anode (cathode) may choose this answer. This answer would be the correct answer if an event occurred that overall reduced the output (such as a very low detector voltage or loss of argon gas pressure).

D. This option is incorrect because the downscale rod block would be less conservative. With the additional argon gas the downscale rod block would occur at a lower neutron level than that of the other detectors so as neutron flux lowers the downscale for IRM A is delayed. A candidate may believe that, because there is more argon gas pressure, the raised detector output makes all the associated actions more conservative and would therefore choose this option.

Technical Reference(s): UFSAR, Volume III, Section VII, Subsection 5.5, IRM
(Attach if not previously provided) IOP 4.1.2, IRM Subsystem, Rev. 23
(including version/revision number) ____

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR001-10-02

5. Describe how changes in each of the following affect detector sensitivity:

b. Detector gas pressure

Question Source: Bank # ____
Modified Bank # 23352 (See attached)
New ____

Question History: Last NRC Exam CNS 2008

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (5)

Comments:

LOD 3

Original Bank question:

Question Number	New, Modified or Bank	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
NRC RO 35	Bank 23352	02		01/30/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	1	1	2	Multiple Choice	

Topic Area	Description
Systems	COR0021202001060D Intermediate Range Monitor

Related Lessons
COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives
COR0021202001060D Given a specific IRM malfunction, determine the effect on any of the following: Reactor power indication

Related References
10CFR55.41(b)6

Related Skills (K/A)	
215003.K5.01	Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: (CFR: 41.5 / 45.3) Detector operation (2.6/2.7)

QUESTION: NRC RO 35

An I&C Tech inadvertently lowers the high voltage supply to an **IRM detector** by half; what is the affect on IRM indication for that channel?

a. Indicated power goes up.

b. Indicated power oscillates.

c. Indicated power goes down.

d. Indicated power does not change.

ANSWER: NRC RO 35

c. Indicated power goes down.

Explanation:

As the I&C Tech lowers the high power supply to the detector, the effectiveness of the detector decreases. This decrease in effectiveness will be indicated as a decreased output on its meter.

Distractors:

a. is incorrect because indicated power goes down but a candidate that believes the IRM functions similar to most other gas filled detectors would choose this answer.

b. is incorrect because indicated power goes down. The candidate that believes the short circuit fluctuations would be equivalent to power would choose this answer.

d. is incorrect because indicated power goes down. The candidate that believes the circuit would fail as is would choose this answer.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215004 K6.01	
Importance Rating	3.2	

215004 Source Range Monitor – Knowledge of the effect that a loss or malfunction of the following will have on the source range monitor (SRM) system: (CFR: 41.7) K6.01 RPS

Question: 36

The plant is performing a reactor startup with the SRM Shorting Link Switches in the **OPEN** position for testing. The following occurs:

- Following an electrical transient during maintenance, contacts in both Reactor Manual Scram Trip Logic "A3" trip relays are welded closed.

How does this failure impact the interaction between SRM channels and RPS?

A. Only SRMs B and D upscale trips can cause a **full** Reactor Scram.

B. Only a half scram from "A" RPS trip system occurs on **any** SRM upscale trip.

C. Only a half scram from "B" RPS trip system occurs on **any** SRM upscale trip.

D. RPS causes a **full** Reactor Scram in the Non-Coincident mode on **any** SRM upscale trip.

Answer:

C. Only a half scram from "B" RPS trip system occurs on **any** SRM upscale trip.

Explanation:

The shorting link switches being OPEN allow the neutron monitoring non-coincident trips in the A3 and B3 scram logics. If the RPS A3 trip relay contacts fail closed, the A3 channel will not relay the SRM upscale high-high trip and prevent the A RPS trip from occurring. This failure will not impact the ability of the SRM channel to input a trip signal. The candidate would choose this answer based on the shorting links being open would cause the A3 RPS Manual Scram Trip logic and B3 RPS Manual Scram Trip Logic to become a two-out-of-two taken once logic. By removing the ability of the A3 logic to trip, a ½ scram could not occur on the A side and a full scam due to SRM inputs would not be possible. This question is a K/A match because the RPS failure is the relay fingers being welded closed. This failure does not allow the contacts to open in the A3 logic when one SRM upscale trip occurs. If the contacts were not welded, one SRM upscale trip causes both A3 and B3 logics to trip which is a full reactor scram.

A. This option is incorrect because the RPS failure prevents any and all SRM induced full Reactor scrams (but not half scrams). The candidate could choose this distractor if he/she did not know the relationship between RPS Manual Scram Trip logic and the non-coincident neutron monitoring trips. Since the RPS failure is on the A it is a common misconception that this failure would only prevent SRM A and C from initiating a full scram.

B. This option is incorrect because the failure present in RPS actually prevents an RPS A trip. The candidate who does not understand that the trip condition of the A3 logic is de-energized would choose this option.

D. This option is incorrect because the failure prevents any and all SRM induced full Reactor scrams. The candidate could choose this distractor if he/she did not understand the non-coincident input of the SRMs through the shorting links or does not recall the Shorting Links switch positions. This answer is plausible because a non-coincident scram from the SRMs does occur on the upscale trip of any SRM.

Technical Reference(s): ______
(Attach if not previously provided) 791E256 Reactor Protection System Elementary Electrical
(including version/revision number) ______

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-30-02, Source Range Monitor
8. Predict the consequences a malfunction of the following would have on the SRM system:
   a. RPS (including shorting switches)

Question Source:	Bank #	23354
	Modified Bank #	________
	New	________
Question History:	Last NRC Exam	________
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 4

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215005 A1.03	
Importance Rating	3.6	

215005 APRM / LPRM – Ability to predict and/or monitor changes in parameters associated with operating the average power range monitor/local power range monitor systems controls including: (CFR: 41.5) A1.03 Control rod block status

Question: 37

The reactor is operating at 15% rated power when the following occurs:

- IRMs "B" and "H" both fail upscale.

Then, before any operator action is taken:

- APRMs "B" and "E" both fail downscale.

What **minimum** action(s) clear **all** rod blocks and/or scrams?

A. Bypass APRM "B" **only**.

B. Bypass APRM "E" **only**.

C. Bypass APRM "B" and IRM "B".

D. Bypass APRM "B" and APRM "E".

Answer:

D. Bypass APRM "B" and APRM "E".

Explanation:

The upscale IRM B and the downscale APRM B together generates an RPS trip on the "B" RPS. APRM "B" or APRM "E" failed downscale cause a rod block.

To clear the RPS trip APRM "B" OR IRM "B" must be bypassed. To clear the rod block, both APRM "B" AND APRM "E" must be bypassed.

Bypassing both APRM "B" and APRM "E" will clear the rod block, and bypassing APRM "B" also clears the RPS trip signal.

Distracters:

A. This option is incorrect because this action will clear the half scram but will not clear the rod block. The candidate who did not understand that a rod block is generated by **EITHER** APRM B **OR** APRM E failed downscale would choose this option. This answer is plausible because the APRM can be bypassed and its direct trip is also bypassed.

B. This option is incorrect because this action will not clear either the rod block or the half scram. The candidate could choose this distractor if he/she did not understand that a rod block is generated by **EITHER** APRM B **OR** APRM E failed downscale and he/she did not know the ½ scram was generated by the upscale IRM B and the downscale APRM B. This answer is plausible because the APRM can be bypassed and its direct trip is also bypassed.

C. This option is incorrect because this action will clear the half scram but will not clear the rod block. The candidate could choose this option if he/she did not know the ½ scram was generated by both the upscale IRM B and the downscale APRM B. This answer is plausible because the APRM and IRM can be bypassed and their direct trips are also bypassed.

Technical Reference(s): IOP 4.1.3, Average Power Range Monitoring System, Rev. 25

Proposed references to be provided to applicants during examination: ___None________

Learning Objective:

Per COR002-01-02, Average Power Range Monitor

5. Describe the interrelationships between the Average Power Range Monitor system and the following:
   a. Reactor Protection System (RPS)
   b. Intermediate Range Monitoring system (IRM)
   f Reactor Manual Control System (RMCS)

Question Source:	Bank #	_23450_
	Modified Bank #	_______
	New	_______
Question History:	Last NRC Exam	_CNS 2006__
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 _(5)_	

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	217000 A2.16	
Importance Rating	3.5	

217000 RCIC – Ability to (a) predict the impacts of the following on the reactor core isolation cooling system (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5) A2.16 Low condensate storage tank level

Question: 38

Given the following conditions:

- RCIC is injecting to the reactor at 400 gpm.
- RCIC suction is from the ECST.
- Torus temperature is 125°F and steady.
- ECST level is 25 inches and lowering.
- Torus level is 12' 9" and rising.

(a) If the conditions persist, how is RCIC impacted?
(b) What action is required next?

	(a) RCIC Suction aligns to torus…	(b) Action required
A.	at ≥23" ECST Level.	Secure RCIC
B.	at ≥23" ECST Level.	Makeup to the ECSTs
C.	at ≤13' 1" Torus Level .	Secure RCIC
D.	at ≤13' 1" Torus Level .	Makeup to the ECSTs

Answer:

B. at ≥23" ECST Level. Makeup to the ECSTs

Explanation:

The alternate source of water for the RCIC pump is the Suppression Pool (torus). This source of water is used if the Emergency Condensate Storage Tank levels are low. The valve (MO-41) for RCIC pump suction from the Suppression Pool will automatically open on a low level of ≥23" from the bottom of either Emergency Condensate Storage Tank. Per ARP 9-4-1/F-2, when the suction transfer alarm is received, action should be taken to provide makeup to the ECST per Procedure 2.2.7, Condensate Storage and Transfer System.

Distracters:

A. This option is incorrect because there is no need to secure the RCIC system following the swap. If the torus temperature were 145°F instead of 125°F, RCIC would need to be secured in order to prevent overheating the lube oil. The candidate who understands the RCIC suction swap but who does not remember the maximum torus temperature that supports RCIC lube oil cooling would choose this answer.

C. This option is incorrect because the RCIC suction swap does not occur on high torus level and the torus water temperature is not high enough to prohibit RCIC operation. The HPCI system does have a suction swap on high level, as well as the low ECST, so a candidate could easily confuse which system has only one parameter that swaps the suction and what that condition causes the swap. If the torus temperature were higher as well the RCIC system would need to be secured due to high oil temperature concerns.

D. This option is incorrect because the RCIC suction swap does not occur on high torus level. The HPCI system does have a suction swap on high level, as well as the low ECST, so a candidate could easily confuse which system has only one parameter that swaps the suction and what that condition causes the swap.

Technical Reference(s):
(Attach if not previously provided) SOP 2.2.67.1,Reactor Core Isolation Cooling Ops, Rev. 31
(including version/revision number) ARP 9-4-1/F-2, Rev. 51

Proposed references to be provided to applicants during examination: None

Learning Objective:
COR002-18-02, Reactor Core Isolation Cooling

11. Describe the interrelationship between RCIC system and the following:

    h. ECSTs

8. Describe the RCIC system design features and/or interlocks that provide for the following:

    a. Alternate water supplies

10. Predict the consequences of the following on the RCIC system:

    c. Low ECST level

11. State the reason for the following:

    b. RCIC suction transfer on Low ECST water level

12. Given plant conditions, determine if the following RCIC actions should occur:

    c. ECST suction transfer

Question Source:	Bank #	______
	Modified Bank #	______
	New	X
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (5)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 A3.02	
	Importance Rating	3.6	

218000 ADS – Ability to monitor automatic operations of the automatic depressurization system including: (CFR: 41.7) A3.02 ADS valve tail pipe temperatures

Question: 39

Automatic initiation of ADS occurs.

What provides positive indication that each ADS valve is open?

A. The red light for each ADS valve is illuminated on Panel 9-3.

B. The green light for each ADS valves is extinguished on panel 9-3.

C. ADS TIMERS ACTUATED, 9-3-1/A-1, and ADS AUX RELAYS ENERGIZED, 9-3-1/B-1 are **both** in alarm.

D. Temperature readings 285ºF to 300ºF for each ADS valve on temperature recorder MS-TR-166 on panel 9-21.

Answer:

D. Temperature readings 285ºF to 300ºF for each ADS valve on temperature recorder MS-TR-166 on panel 9-21.

Explanation:
When an SRV lifts, the tailpipe temperature and pressure increase to approximately 285-300ºF and 30 psig in a constant enthalpy process that superheats the SRV exhaust steam. The temperature recorder provides a positive indication that the ADS SRVs are open. The six red lights associated with determine that the solenoids for the ADS valves are energized. They do not indicate SRV position however as the valve may fail to open even with the solenoid energized. The six green lights extinguish on an ADS signal, but do not indicate SRV position.

The annunciators are tied to the ADS logic being satisfied. They do not indicate SRV position.

The SRV tailpipe temperatures actually determine steam flow from the SRV, hence they indicate that the individual SRVs are open and passing steam.

A. This option is incorrect because the red lights only indicate that the solenoids are energized but do not provide positive indication that the valves are open. However the red lights associated with each ADS valve are energized every time ADS automatically actuates. A candidate would see the red lights illuminated whenever he/she sees an ADS actuation in the simulator. And while this does occur for every valve it is not positive indication of valve position.

B. This option is incorrect because the green lights extinguishing only indicate that the ADS logic is satisfied. It does not provide positive indication of SRV position. However, this occurs with every ADS actuation so a candidate who does not completely understand the system may associate that indication with positive indication of valve position and choose this option.

C. This option is incorrect because the alarms are an expected indicator for ADS logic met. They do not indicate SRV position. These alarms do occur for every ADS actuation and are a focal point for the operator during transient/accident conditions which is why a candidate may choose this option.

Technical Reference(s): AP 2.4SRV, Stuck Open Relief Valve, Rev. 15

Proposed references to be provided to applicants during examination: __None__________

Learning Objective:

Per COR002-16-02, Nuclear Pressure Relief
6. Briefly describe the following concepts as they apply to NPR:

d. Tail pipe temperature monitoring

Question Source:	Bank #	_______	
	Modified Bank #	_1860__	(See attached)
	New	_______	
Question History:	Last NRC Exam	____________	
Question Cognitive Level:	Memory or Fundamental Knowledge	_____	
	Comprehension or Analysis	__X__	
10 CFR Part 55 Content:	55.41 _(7)_		

Comments:

LOD 3

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	1860	00	08/09/1999	06/15/2005	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021602001040A, COR0021602001050H Nuclear Pressure Relief

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001040A	Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Tail pipe temperatures
COR0021602001050H	Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: Detection of valve leakage

Related References
NONE

Related Skills (K/A)	
239002.A3.03	Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: (CFR: 41.7 / 45.7) Tail pipe temperatures (3.6/3.6)

QUESTION: 29 1860 (1 point(s))

Following testing of the SRVs, it could be assumed that one would be leaking based on which of the following indications?

a. Temperature recorder indicating between 270°F and 300°F.

b. White light on suspect SRV would be illuminated.

c. Tailpipe temperature 150°F to 220°F indicated.

d. Computer indication of tailpipe temperatures 150°F to 220°F.

ANSWER: 29 1860

a. Temperature recorder indicating between 270°F and 300°F.

REFERENCE: Nuclear Pressure Relief Text

Distractors:
b. Amber light illuminated when pressure switch picked up.
c. Tailpipe temperature would indicate between 270°F and 300°F.
d. Tailpipe temperature would indicate between 270°F and 300°F.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A4.06	
	Importance Rating	3.6	

223002 PCIS/Nuclear Steam Supply Shutoff – Ability to manually operate and/or monitor in the control room: (CFR: 41.7) A.4.06 Confirm initiation to completion

Question: 40

Following a transient the reactor is scrammed and the mode switch is place to shutdown. The following conditions are present:

- RPV pressure falls to 600 psig then starts slowly rising.
- RPV water level falls to -42 inches and then rises.
- Drywell pressure rises to 1.2 psig and stabilizes.
- All MSIVs are open.
- RHR-MO-920, AOG Steam Supply valve, is open.
- RR-AO-741, Reactor Water Sample Isolation valve, is open.
- RCIC-MO-15, Steam Supply Line Isolation valve, is open.

Which valve(s) is/are required to be CLOSED to ensure PCIS initiation is complete?

A. MSIVs

B. RHR-MO-920

C. RR-AO-741

D. RCIC-MO-15

Answer:

B. RHR-MO-920

Explanation:
The reactor water level is below the level (+3 inches) that initiates a Group 2 isolation. When a Group 2 isolation occurs **RHR-MO-920, AOG STEAM SUPPLY VLV is required to close and if the isolation does not complete the valve closure the operator is to close the valve.** No conditions other than those that would result in the initiation of a Group 2 isolation have occurred.

Distracters:

A. This option is incorrect because RPV level did not get low enough for the Group 1 Isolation nor did steam pressure go low enough with the mode switch in RUN to cause a Group 1 isolation. The candidate who does not recall the water level for the Group 1 isolation or who fails to analyze the effect of the mode switch position may choose this answer. If the mode switch were not in SHUTDOWN then the MSIVs would be required to be closed. This option would be correct if the mode switch were in RUN. This answer is plausible because MSIVs will close on a low RPV level.

C. This option is incorrect because RPV level did not get low enough for the Group 7 Isolation. Water level would have to lower to ≥-113" for this isolation to occur. A candidate may confuse the Group 7 isolation with a Group 6 isolation which would occur at the water level provided and would therefore choose this option.

D. This option is incorrect because reactor pressure is not low enough to cause a Group 5 isolation. The candidate who is unsure of the Group 5 isolation setpoint on reactor pressure may choose this option particularly if they confuse the isolation and initiation conditions as a RCIC initiation signal is present. This answer is plausible because the listed valve closes on a low RPV pressure signal.

Technical Reference(s): GOP 2.1.22, Recovery From a Group Isolation, Rev. 59

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR002-03-02, Containment
21. Given plant conditions, determine if the following should have occurred:
    a. Any of the PCIS group isolations

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	239002 2.2.12	
	Importance Rating	3.7	

239002 SRVs – 2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Question: 41

While performing SRV Surveillance Test 6.ADS.201 during startup following a refuel outage, the BOP Operator reports that SRV RV-71 E Control Switch has been placed to OPEN.

When conditions stabilize, which of the following indications validate that SRV RV-71 E is full open?

A. Main Generator output lowers.

B. Total indicated steam flow rises.

C. A reduction of Total Feedwater Flow.

D. Bypass valves throttle in the closed direction.

Answer:

D. Bypass valves throttle in the closed direction.

Explanation:
The acceptance criteria for 6.ADS.201 specify the valid parameters for verifying that an SRV has properly opened. The operability limit specified is a change of BPV Position ≥2%.

Distracters:

A. This option is incorrect because the Main Generator is off-line during SRV testing following a refuel outage. This is plausible if the candidate only thinks of total plant effect without realizing the Main Generator is off-line.

B. This option is incorrect because total steam line flow is not an approved method of verifying SRV position for surveillance purposes. This answer is plausible because the SRVs are located upstream of the main steam line flow measurement devices, therefore total indicated steam flow will actually lower. The candidate who selects this answer could have an incorrect mental model of steam flow measurement during SRV surveillance and not remember that total steam line flow is not used.

C. This option is incorrect because total feedwater flow will actually rise. This occurs to make up for the lost inventory from the SRV to the Suppression Pool. This is plausible if the candidate has an incorrect mental model of feedwater response with an open SRV.

Technical Reference(s): 6.ADS.201 ADS Manual Valve Actuation Rev.11

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-16-02, Nuclear Pressure Relief

4. Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters:
   - c. Reactor pressure
   - f. Reactor power
   - g. Turbine load

Question Source:	Bank #	______	
	Modified Bank #	______	
	New	X	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge		____
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41 (7)		

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K1.07	
	Importance Rating	2.6	

259002 Reactor Water Level Control – Knowledge of the physical connections and/or cause-effect relationships between reactor water level control system and the following: (CFR: 41.2) K1.07 Rod worth minimizer: Plant-Specific

Question: 42

In order to determine its mode of operation, what information must the Rod Worth Minimizer (RWM) receive from the Reactor Vessel Level Control System (RVLCS)?

A. Total main steam and feedwater flow rates.

B. Total main steam flow and feedwater flow rate mismatch.

C. Time that total main steam flow and feedwater flow are above limits.

D. Reactor power level calculated from total main steam and feedwater flow rates.

Answer:

A. Total main steam and feedwater flow rates.

Explanation:
The RWM operates during plant startup or shutdown. Based upon steam flow and feedwater flow rates, the mode of operation of the RWM is determined. The algorithms come from the Reactor Vessel Level Control System and determine when the RWM starts or stops enforcing predetermined control rod movements. When operating in the Low Power Alarm Point (LPAP) mode, the RVLCS sends the total main steam flow signal to the RWM. The RWM algorithm determines if the total main steam signal has been above 35% for 60 seconds. If these conditions are met, then operation is above the LPAP. Operation in the Transition Zone (TZ) is operation between LPSP and LPAP. Occurs when > 20% total Main Steam flow AND > 20% total feedwater flow (with each condition present for at least 60 seconds) AND ≤ 35% total Main Steam flow (for any amount of time) as sensed by the LPSP and LPAP algorithms in the Reactor Vessel Level Control System, respectively. LPSP is a variable used by the RWM program and the RWM mode will be OPERATING < LPSP when either total Main Steam flow is at or below 20% ***or*** total feedwater flow is at or below 20% (this condition being determined by the LPSP algorithm in the RVLCS program) for any period of time.

Distracters:

B. This option is incorrect because the RWM does not use main steam flow and feedwater flow mismatch for determining its operating mode. This answer is plausible because the RVLCS does use the mismatch for its algorithms and the candidate may confuse the two.

C. This option is incorrect because the time measurement comes from the RWM algorithm and not the RVLCS. This option is plausible because main steam and feedwater flow rates are timed to determine operating mode. The candidate who does not know which system is measuring the time that flows are at a given level would select this option.

D. This option is incorrect because reactor power is not an input to the RWM. This option is plausible because the RWM is required to be in service when below 9.85% rated power. The candidate who knows the Technical Specification requirements for BPWS and knows the reactor power level where BPWS constraints must be met would select this answer.

Technical Reference(s): IOP 4.2, Rod Worth Minimizer, Rev. 29

Proposed references to be provided to applicants during examination: __None___________

Learning Objective:

Per COR002-32-02, Reactor Vessel Level Control

2. Describe the interrelationship between RVLC and the following:
   j. RWM

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	____________
Question Cognitive Level:	Memory or Fundamental Knowledge	__X__
	Comprehension or Analysis	_____
10 CFR Part 55 Content:	55.41 _(7)_	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K3.02	
	Importance Rating	3.6	

261000 SGTS – Knowledge of the effect that a loss or malfunction of the standby gas treatment system will have on the following: (CFR: 41.7) K3.02 Off-site release rate

Question: 43

An accident occurred that resulted in fuel failure and a breach of the reactor coolant system boundary. The “A” SGT train is in service to support Primary containment venting when the following alarms:

- Annunciator K-1/A-2 SGT A HIGH MOISTURE

What SGT system component is primarily affected and how is the offsite release rate affected?

A. The Charcoal Filter and release rate rises primarily due to iodine activity.

B. The Charcoal Filter and release rate rises primarily due to particulate activity.

C. The High Efficiency Inlet Filter and the release rate rises primarily due to iodine activity.

D. The High Efficiency Inlet Filter and the release rate rises primarily due to particulate activity.

Answer:

A. The charcoal filter and release rate rises primarily due to iodine activity.

Explanation:
The quantity of airborne contaminants is reduced as the air passes through the SGT train. This is accomplished by both mechanical filtration in the filters (both rough and HEPA type) and by chemical adsorption in the charcoal filter (iodine being the preeminent isotope of concern).

Each train is equipped with an electric air heating element, which reduces the relative humidity of the air prior to the charcoal filter, thus improving the efficiency of the charcoal filter. If the moisture content of the air stream becomes too high, then the moisture is carried on to the charcoal filter. The charcoal filter becomes less efficient for adsorbing the iodine.

Distracters:

B. This option is incorrect because although the primary SGT system component that is affected is the charcoal filter. The charcoal filter needs the SGT heater to ensure that the relative humidity of the gas stream entering the charcoal filter is sufficiently low to allow the charcoal filter to function efficiency with a relatively high adsorption rate. As the humidity levels rise, the efficiency of the charcoal to adsorb iodine diminishes. A candidate who does not understand that the charcoal is primarily for iodine may choose this answer.

C. This option is incorrect because although iodine would be a concern with the high humidity the high efficiency filter is capable of performing its function with high humidity. A candidate who confuses the primary function of the high efficiency inlet filter and the charcoal filter would choose this option.

D. This option is incorrect because component affected is the charcoal filter not the inlet filter and the primary concern for release is the iodine. A candidate who believes that the inlet filter efficiency is affected by the humidity may choose this option.

Technical Reference(s): USAR XIV, Section 8.2
Procedure 2.2.73, Standby Gas Treatment System, Rev. 52.

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR002-28-2, Standby Gas Treatment System

7. Given a specific Standby Gas Treatment System malfunction, determine the effect on any of the following:
   b. Off-site release rate

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262001 K2.01	
Importance Rating	3.3	

262001 AC Electrical Distribution – Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Off-site sources of power.

Question: 44

What is a power source to the Emergency Service Station Transformer?

A. Directly from the Auburn line.

B. Directly from the Cornfield Substation.

C. 161 KV substation via the T6 transformer.

D. 161 KV substation via the T7 transformer.

Answer:
C. 161 KV substation via the T6 transformer.

Explanation:
During normal station operation, the Emergency Service Station Transformer (ESST) is energized by the 69 kV transmission line from the 69 kV Bay of the 161 kV Substation through Air Break Switch 5298. The Emergency Transformer supply can be aligned to either the Cooper 161 kV System via Transformer T6 or to the OPPD 69 kV line.

Distracters:

A. This option is incorrect because the Auburn line connects with the 161 kV Substation. From the 161 kV Substation the power must go through the 69 kV Bay and the T6 transformer to connect with the ESST. The Auburn line does not connect directly with the ESST. The 161 kV switchyard has recently gone through a major design change and the candidate may not fully understand the new configuration. This answer is plausible because the Auburn line does supply the ESST, just not directly.

B. This option is incorrect because the 69 kV Cornfield Substation has been removed during the 161 kV Substation major design change. The candidate who recalls the old arrangement would select this answer. This answer is plausible because the Cornfield Substation was previously used to directly feed the ESST.

D. This option is incorrect because the T7 transformer does not feed the ESST. The T7 transformer is part of the AC distribution system and is a new addition to the station so the candidate who confuses it with the T6 transformer would select this answer.

Technical Reference(s): SOP 2.2.17, ESST, Rev. 64

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR001-01-01, OPS AC Distribution

7. State the electrical power supplies to the following:
    a. Off-Site Sources of Power

Question Source:	Bank #	____
	Modified Bank #	____
	New	X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	____

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262002 K4.01	
Importance Rating	3.1	

262002 Uninterruptible Power Supply (A.C. / D.C.)
Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including:

K4.01Transfer from preferred to alternate source.

Question: 45

The plant is in a normal full power electrical lineup. The following alarm is received:

C-4/E-7 NO BREAK SYSTEM INVERTER 1A VOLTAGE FAILURE.

What is the current source of power to the NBPP?

A. MCC-L via a step down transformer and the inverter cabinet static switch.

B. MCC-R via a step down transformer and the inverter cabinet static switch.

C. MCC-L via a step down transformer and then directly to the power panel.

D. MCC-R via a step down transformer and then directly to the power panel.

Answer:

B. MCC-R via a step down transformer and the inverter cabinet static switch.

Explanation:
System Operating Procedure 2.2.22, describes NBPP automatic transfer to MCC-R. See Attachment 1 Steps 1.2.5 and 2.3.

Power to the No-Break Power Panel (NBPP) #1 is normally supplied from 250 VDC bus 1A through inverter 1A and a static switch. The inverter failure alarm indicates that the power into or out of the inverter is failed which causes the NBPP to transfer to MCC-R. MCC-R powers the NBPP through a step down (115V AC) transformer to the static switch in the inverter cabinet.

The static switch can also be operated with the NBPP PWR TRANSFER switch on Panel C (MCC or IVTR) or by pressing the ALTERNATE SOURCE SUPPLYING LOAD or INVERTER SUPPLYING LOAD button on the inverter. The NBPP power can also be transferred by placing the MANUAL BYPASS SWITCH on the inverter to ALTERNATE SOURCE TO LOAD or

NORMAL OPERATION per SOP 2.2.22. The static switch and manual bypass switch transfer the NBPP power supply in a make before break logic.

Distracters:

A. This answer is incorrect because MCC-L powers the PMIS-UPS inverter as an alternate supply. This answer is plausible because the PMIS-UPS is a different uninterruptable power supply at the station. The candidate who confuses the NBPP and PMIS-UPS panels would select this answer.

C. This answer is incorrect because MCC-L powers the PMIS-UPS inverter as an alternate supply. This answer is plausible because the PMIS-UPS is a different uninterruptable power supply at the station. The candidate who confuses the NBPP and PMIS-UPS panels would select this answer.

D. MCC-R automatically powers NBPP through the Static switch. To feed the NBPP directly requires a MANUAL BYPASS SWITCH to be manipulated at the inverter cabinet. This answer is plausible because powering NBPP by bypassing the static switch is a means of powering the panel. The candidate who cannot recall the different configuration arrangements in the inverter cabinet would select this answer.

Technical Reference(s): System Operating Procedure 2.2.22 Rev 70
B&R Electrical Drawing Sheet 16 (Rev N25)

Proposed references to be provided to applicants during examination: None

Learning Objective: COR0010102 AC Electrical Distribution

COR0010102001090G Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Transfer from preferred power to alternate power supplies

Question Source:	Bank #	25667
	Modified Bank #	
	New	

Question History: Last NRC Exam 2012

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K5.01	
	Importance Rating	2.6	

263000 DC Electrical Distribution – Knowledge of the operational implications of the following concepts as they apply to D.C. electrical distribution: (CFR: 41.5) K5.01 Hydrogen generation during battery charging.

Question: 46

The plant is operating at 100% power with the following condition:

- Battery charge in progress following the replacement of a cell in the Division II 250 VDC station battery.
- Complete loss of Battery room ventilation occurs.

What is the operational concern for the present conditions?

A. Hydrogen buildup in the battery room is a fire hazard.

B. Hydrogen buildup in the battery room displaces oxygen.

C. Battery room temperature rise causes cell overheating and loss of electrolyte level.

D. Battery room temperature rise leads to cell reversal conditions in the new replacement battery cell.

Answer:
A. Hydrogen buildup in the battery room is a fire hazard.

Explanation:
Battery room ventilation is required to maintain room temperature and disperse hydrogen generated from battery charging. In the case of a battery charge in progress, the hydrogen removal function is the operational concern. The hydrogen buildup is a fire/explosive hazard and concern. Although hydrogen can displace oxygen in a space, it becomes a fire hazard at much lower concentrations.

The candidate should understand that hydrogen removal is the concern and that the buildup of hydrogen is a fire/explosion hazard.

Distracters:

B. This option is incorrect because even though hydrogen buildup is a concern in the battery room, it is the fire/explosive hazard that is the concern and not displacement of oxygen as an explosive or fire hazard would exist for a significant period of time before displacement of sufficient oxygen to cause a problem could occur, if at all. Since operators do deal with confined spaces and habitable environments the operator could believe that the battery room fan is there to prevent displacement of oxygen.

C. This option is incorrect because even though battery room temperature may rise the concern with the loss of ventilation is not the temperature but the hydrogen. But because there could be a room temperature rise associated with the loss of ventilation the candidate may believe that the reason is temperature and a high rate of electrolyte loss.

D. This option is incorrect because even though battery room temperature may rise the concern with the loss of ventilation is not the temperature but the hydrogen. A candidate may not know the contributory factors associated with cell reversal but may know that it is a serious battery operational concern. That candidate may also believe that high cell temperature could contribute to cell reversal and would choose this option.

Technical Reference(s): ____________________
(Attach if not previously provided) SOP 2.2.24.1, 250VDC Electrical System (Div. 1)
(including version/revision number) ____________________

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-07-02, DC Electrical Distribution

10. Briefly describe the following concepts as they apply to DC Electrical Distribution System.
    a. Hydrogen generation during battery charging.

Question Source:	Bank #	______
	Modified Bank #	21329 (See attached)
	New	______
Question History:	Last NRC Exam	CNS 2005
Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (5)	

Comments:

LOD 2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	21329	00	08/15/2005	10/01/2005	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020702, Predict the impact of a loss of ventilation on the battery and based on that prediction determine actions to mitigate the impact.

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives
NONE

Related References	
(B)(7)	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(B)(8)	Components, capacity, and functions of emergency systems.

Related Skills (K/A)	
263000.A2.02	Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the conseq...: (CFR: 41.5 / 45.6) Loss of ventilation during charging (2.6/2.9)

QUESTION: 49 21329 (1 point(s))

The plant is operating at 100% power with the following condition:

- Battery charge in progress following the replacement of a cell in the **Division II** 125/250 VDC station battery.
- EF-C-1C, BATT RM EXH FAN is out of service due to a motor failure.

MCC-LX is lost.

What impact does this have on the batteries?

What action is required?

a. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Restore a battery room fan to service within 7 days.

b. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Install and operate portable ventilation equipment for the battery rooms.

c. Power is lost to the Division II Battery Charger with eventual loss of battery voltage. Align the Division II battery charge to its alternate supply.

d. Power is lost to the Division II Battery Charger with eventual loss of battery voltage. Declare the Division II Battery inoperable.

ANSWER: 49 21329

b. Battery room ventilation is lost with possible hydrogen buildup in the battery room. Install and operate portable ventilation equipment for the battery rooms.

The loss of MCC-LX results in the loss of the only available Battery room ventilation fan.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K6.03	
	Importance Rating	3.5	

264000 EDGs – Knowledge of the effect that a loss or malfunction of the following will have on the emergency generators (diesel): (CFR: 41.7) K6.03 Lube oil pumps

Question: 47

DG1 is manually started for post maintenance testing:

- Generator output breaker is open.
- Engine driven oil pump fails (no oil flow).

What condition automatically trips the diesel generator?

A. Low Lube Oil Pressure

B. High Lube Oil Temperature

C. Low Turbocharger Oil Pressure

D. High Thrust Bearing Oil Temperature

Answer:

A. Low Lube Oil Pressure

Explanation:
When the diesel generator is manually started all the diesel generator trips are in effect. With the loss of the engine driven oil pump and the loss of all lube oil pressure, the diesel generator trips at <20 psig lube oil pressure.

B. This option is incorrect because the diesel generator does not trip on high oil temperature, with the loss of lube oil pressure a low oil pressure trip would occur before engine oil temperature became elevated. This option is plausible if a candidate does not know the diesel generator trips but reasons that without lube oil temperatures would elevate and trip the diesel.

C. This option is incorrect because the turbocharger oil pressure trip occurs at a lower pressure than does the engine oil pressure trip. So as oil pressure falls the engine oil trip point would be reached first. But a candidate who does not know the relative pressures or the configuration of the oil supply to the turbocharger may choose this option.

D. This option is incorrect because this condition would occur after the oil pressure had tripped the diesel generator. But the loss of lube oil could cause the turbocharger thrust bearing temperature to elevate so a candidate could reason that this would trip the diesel generator.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)   SOP 2.2.20, Standby AC Power System (Diesel Generators). Rev. 92

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-08-02, Diesel Generators

11. Predict the consequences a malfunction of the following would have on the Diesel Generators:
    c. Lube Oil pumps

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 A3.02	
	Importance Rating	2.9	

300000 Instrument Air – Ability to monitor automatic operations of the instrument air system including: (CFR: 41.7) A3.02 Air Temperature

Question: 48

Air compressors are operating with the Compressor Control Module (CCM) in LOCAL and the lineup is:

- Compressor 1A is in Lead and running.
- Compressor 1B is First Backup and is idle.
- Compressor 1C is Second Backup and is idle.

Reactor Equipment Cooling (REC) to the air compressors is lost due to an REC pipe rupture.

What condition automatically trips Compressor 1A?
What automatically occurs following the trip of Compressor 1A?

A. high air temperature
Compressor 1B starts and continuously supplies air loads.

B. high air temperature
Compressor 1B starts but trips soon after it starts.

C. low cooling water pressure
Compressor 1B starts and continuously supplies air loads.

D. low cooling water pressure
Compressor 1B starts but trips soon after it starts.

Answer:

A. high air temperature
Compressor 1B starts and continuously supplies air loads.

Explanation:

The normal cooling water lineup is REC to Compressor 1A and TEC to Compressors 1B and 1C. When REC is lost to Compressor 1A, air temperatures will rise until compressor 1A trips. Lowering air pressure will start the 1st Backup compressor (which is cooled by TEC) so it continues to operate supplying system air loads. All air loads should be supplied. Also the sequence with compressors with CCM in LOCAL is lead cycles 110 to 100 psig, 1st Backup starts at 93 psig and 2nd Backup starts at 90 psig.

If power is lost to the REC-TEC cross-tie valves (which is not the case here), then the REC and TEC alignment to the compressors when power is restored is 1A and 1B supplied by REC and 1C supplied by TEC. This response makes choices B and D highly plausible. The compressor protection from loss of cooling water is high temperature trips, not low cooling water pressure.

B. This option is incorrect because B compressor does not trip on high air temperature but continues to operate as its cooling water supply is from TEC. The candidate who knows that compressor A trips due to high temperature but who also believes that compressor B is cooled by REC may choose this answer. This is a likely misconception that candidates could have, as two of the compressors are supplied by one closed cooling water system and one is supplied by a different system.

C. This option is incorrect because the compressors A does not trip due to low cooling water pressure. But many components that require cooling water do trip due to low cooling water pressure so a candidate may choose this option. Air compressor 1B does start and carry the load which makes this option plausible.

D. This option is incorrect because the A compressor does not trip due to low cooling water pressure nor will the 1B compressor trip as it remains supplied with cooling water from the TEC system. The candidate who does not know the air compressor trips and believes that the cooling water realigns and that compressor 1C is the only compressor supplied with cooling water would choose this option.

Technical Reference(s): SOP 2.2.59, Plant Air System. Rev. 74
(Attach if not previously provided) EP 2.5AIR, Loss of Instrument Air, Rev. 19
(including version/revision number) COR001-17-01

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR001-17-01
6 Predict the consequences the following would have on the Plant Air system:
a.REC failure
b.TEC failure

And

10. Given plant conditions, determine if any of the following should occur:
c.Air Compressor automatic trip

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	__X__

10 CFR Part 55 Content: 55.41 _(7)_

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A2.01	
	Importance Rating	3.3	

400000 Component Cooling Water – Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions use procedures to correct, control or mitigate the consequences of those abnormal operations: (CFR: 41.5) A2.01 Loss of CCW pump.

Question: 49

The unit is operating at 50% power with REC heat exchanger B and 3 REC pumps in service.

- One REC pump trips for unknown reasons.
- M-1/A-1, REC SYSTEM LOW PRESSURE alarms.

(1) How does the system respond?
(2) Per the alarm procedure, what action is required?

A. (1) Non-critical loads isolate immediately.
(2) Start another REC pump.

B. (1) Non-critical loads isolate immediately.
(2) Isolate Augmented Radwaste cooling.

C. (1) Non-critical loads isolate after 40 second delay.
(2) Start another REC pump.

D. (1) Non-critical loads isolate after 40 second delay.
(2) Isolate Augmented Radwaste cooling.

Answer:

C. (1) Non-critical loads isolate after 40 second delay.
(2) Start another REC pump.

Explanation:
The REC heat exchanger outlet piping contains pressure switches that isolate the non-critical loads after a 40 second time delay. The M-1/A-1 alarm procedure directs starting additional REC pumps. The REC system contains 4 pumps and with 3 pumps in operation, one pump is available to be started. Starting the REC pump brings system pressure back to the pressure before the pump tripped so no valve isolation is expected to take place.

A. This answer is incorrect because non-critical loads isolate after heat exchanger outlet pressure remains below the isolation setting for 40 seconds. Starting another REC pump is the correct mitigating action to take. The candidate who cannot recall a time delay on the valve isolation would choose this answer. This answer is plausible because the non-critical loads of the REC system do isolate on a system low pressure condition.

B. This answer is incorrect because non-critical loads isolate after heat exchanger outlet pressure remains below the isolation setting for 40 seconds. Isolating Augmented Radwaste cooling is incorrect because this is one of the valves that is designed to close on low system pressure. The candidate who cannot recall a time delay on valve isolation or all the valves that isolate would select this answer. This answer is plausible because the non-critical loads of the REC system do isolate on a system low pressure condition.

D. This answer is incorrect because isolating Augmented Radwaste cooling is not the correct action to take to mitigate the condition. The correct action is to start another REC pump. The system response to isolate after a 40 second time delay is correct. The candidate who recalls the isolation time delay would select this answer. This answer is plausible because the system response is correct and isolating the Augmented Radwaste cooling would help preserve system cooling available to the critical loads.

Technical Reference(s): Procedure 2.3_M-1, Panel M, Rev. 16.

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-19-02

6. Given a specific REC malfunction, determine the effect on any of the following:
   a. REC header pressure
   d. Standby REC pump operation

Question Source:	Bank #	______
	Modified Bank #	______
	New	X
Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (5)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 A1.07	
	Importance Rating	4.3	

211000 SLC – Ability to predict and/or monitor changes in parameters associated with operating the standby liquid control system controls including: (CFR: 41.5) A1.07 Reactor power

Question: 50

The plant operating at 100% rated power with SLC pump 1A out of service when an ATWS occurs:

- No control rods insert.
- The MSIVs are closed.
- Average torus temperature is just below BITT and rising slowly.

SLC pump 1B is placed to START and the following conditions are present:

- Reactor power is 40% and slowly lowering.
- Boron is injecting with an initial tank level of 80%.

Assuming the SLC pump is operating at its design flow rate, what is the approximate SLC tank level and status of the Average Power Range Monitor (APRM) downscales after 20 minutes?

	SLC Tank Level	APRMs
A.	54%	**not** downscale
B.	54%	downscale
C.	26%	**not** downscale
D.	26%	downscale

Answer:

B. 54% downscale

Explanation:
Per COR002-29-02
At the minimum flow rate of 38.2 gpm, each positive displacement pump is capable of injecting the SLC tank contents into the reactor within 82 to 116 minutes (41 to 58 minutes with two), independent of the solution concentration in the tank. Two 100% capacity, positive displacement, stainless steel injection pumps are provided. They are triplex plunger type pumps with a designed flow rate of 53 gpm.

1 SLC pump at 53 gpm injection flow will inject 26% of the SLC tank in approximately 20 minutes.

The candidate should predict that 20 minutes of SLC injection will inject HSBW (drop tank level by 26 percent) which in turn will drop reactor power below 3%.

A. This option is incorrect because, with 26% of the SLC injected, hot shutdown boron weight is injected and so reactor power will be less than 3% (APRM downscale). A candidate who believes that HSBW has not yet been injected (the candidate who confuses HSBW with Cold shutdown boron weight) may choose this answer believing that until 60% of the tank is injected that the APRM downscale will not be in.

C. This option is incorrect because SLC tank level would not be at 26% after 20 minutes with only one pump in operation. This would be the approximate level had two pumps been in operation so a candidate who fails to evaluate that only one pump is in operation may choose this option. Additionally the APRMs would be downscale but a candidate may believe that more than 60% of the tank must be injected in order to get reactor power less than 3%.

D. This option is incorrect because SLC tank level would not be at 26% after 20 minutes with only one pump in operation. This would be the approximate level had two pumps been in operation however so a candidate who fails to evaluate that only one pump is in operation may choose this option.

Technical Reference(s): SOP 2.2.74, Standby Liquid Control, Rev. 52
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per CORR002-29-02
6. Briefly describe the following concepts as they apply to SLC:
   c. Shutdown margin

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A4.03
	Importance Rating	3.6

215003 IRM - Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) - A4.03 IRM range switches

Question: 51

A Plant startup is in progress with reactor power at or near the point of adding heat.

Reactor period is near infinity with the Reactor Mode Select switch in STARTUP / HOT STANDBY position.

IRM C is on Range 7 and indicates 35 (0 - 125 Scale) on the Panel 9-5 Recorder, when its range switch is taken to Range 6 (0 - 125 Scale).

What is the new indication for IRM C and what automatic action(s) occur(s)?

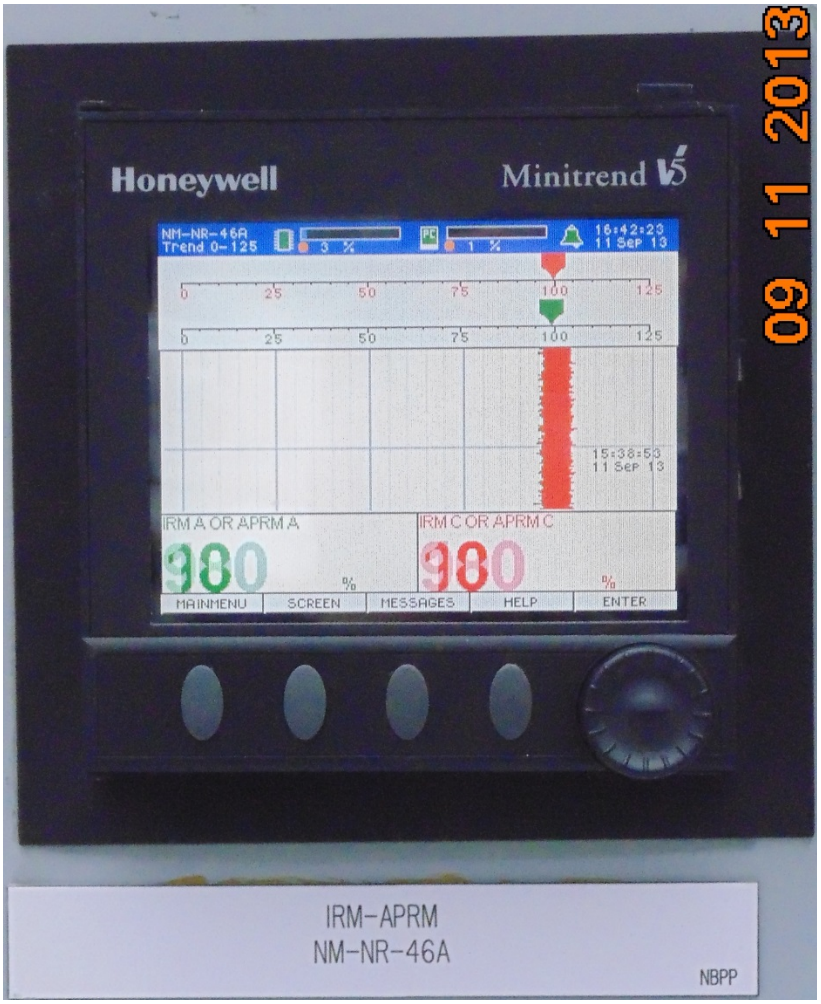
	Recorder Indication	Automatic Action
A.	40	Rod Block and Half Scram.
B.	110	Rod Block Only.
C.	40	Rod Block Only.
D.	110	Rod Block and Half Scram.

Answer:

B. 110 Rod Block Only.

Explanation:

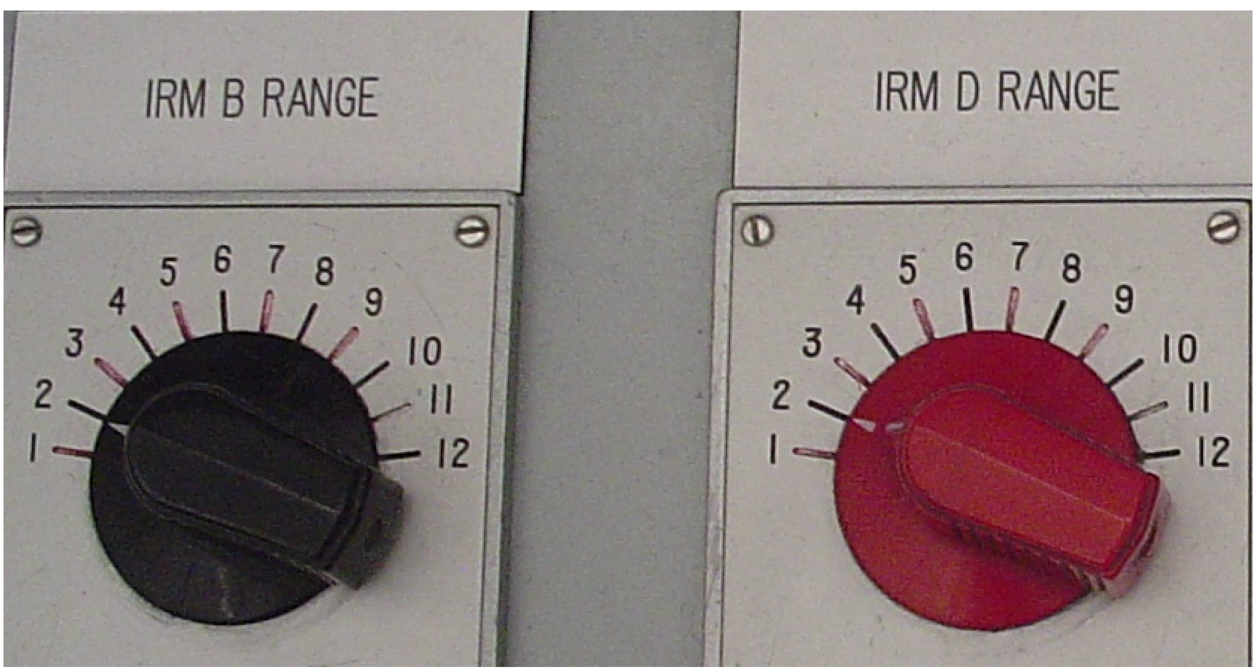
Placing the IRM range switch to the next lower scale will increase the current IRM recorder reading by approximately 3.125. This will result in 35% x 3.125 = 109.4% which is above the TRM Control Rod Block Instrumentation setpoint of ≤108/125 of Full Scale.



Panel 9-5 IRM Recorder example showing 1-125% scale.

The IRM indicator below is located on back panel 9-12 and illustrates the red 0-40% and black 0-125% scale.

IRM Range Switch example. It should be noted that these switches located on panel 9-5 are mechanically pinned to prohibit going above range 9.



Distracters:

A. This option is incorrect because IRM C should increase in value rather than decrease. Additionally, a Rod Block AND Half Scram would not occur at this value of the scale. A value of 40 is at the top end of the lower scale. This option is plausible if the candidate does not remember that ranging down the IRM will cause the indicated value to go up. An IRM range switch manipulation error has recently occurred at CNS.

C. This option is incorrect because IRM C should increase in value rather than decrease. Additionally, no trip or rod block setpoint would be exceeded at 40% of scale. This option is plausible if the candidate does not remember that ranging down the IRM will cause the indicated value to go up.

.

D. This option is incorrect because the IRM C Half Scram Technical Specification value of ≤121/125 is not met. This option is plausible if the candidate did not fully calculate and remember the TS Half Scram IRM value.

Technical Reference(s): IOP 4.1.2 Intermediate Range Monitoring System, Rev. 23
Major Design Change DC-76-1
CNS ESAR

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Intermediate Range Monitor / COR002-12-02

LO-01: State the purpose of the following items related to Intermediate Range Monitoring:

h. Range switch

LO-05: Describe the IRM system design features and/or interlocks that provide the following:

d: Varying system sensitivity levels using Range switches

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 4

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 2.4.3
	Importance Rating	3.7

215004 Source Range Monitor - Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4)

Question: 52

Which one of the following is a Post Accident Monitoring (PAM) instrument?

A. Source Range Monitor (SRM)

B. Traversing In-core Probe (TIP)

C. Condensate Storage Tank Level Indicator

D. Reactor Building Ventilation Exhaust Plenum Radiation Monitors

Answer:

A. Source Range Monitor (SRM)

Explanation:

TLCO 3.3.3 Post Accident Instrumentation (PAM) Instrumentation, Table 3.3.3-1 specifies the instruments that are post accident instrumentation. The neutron monitoring systems in the table are the SRMs, IRMs, and APRMs. The only instrument listed in the options that is a PAM instrument is the SRM.

Distracters:

B. This option is incorrect because the TIP system is not a PAM instrument. Because the TIP can enter the core and provide data for different core locations a candidate could believe that this system function is required post accident and choose this option.

C. This option is incorrect because the Condensate Storage Tank level instrument is not a required PAM instrument. Because the tank level instruments provide control room alarms and the tank is an alternate suction source for low pressure ECCS a candidate may choose this option.

D: This option is incorrect because the reactor building ventilation radiation monitoring instruments are not used for post-accident monitoring. Because other ventilation radiation monitoring instruments (Turbine Building) are PAM required instruments a candidate may choose this option.

Technical Reference(s): TRM

Proposed references to be provided to applicants during examination: none

Learning Objective:

OPS Source Range Monitor/COR002-30-02

LO-02 Given conditions and/or parameters associated with the SRM system, determine if related Technical Specification and Technical Requirements Manual Limiting Conditions for Operation are met.

D. Technical Requirements Manual
   2. T 3.3.3, Non-Type A, Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

INT007-06-02 TRM – Instrumentation

1. Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met:
   c. T 3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring PAM Instrumentation

2. Discuss the applicable Bases associated with each of the following TLCOs:
   c. T 3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring PAM Instrumentation

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (6)	

Comments:

LOD 4

IV. T3.3.3, Non-Type A, Non-Category 1 Post Accident Monitoring (PAM) Instrumentation

EO1c A. TLCO 3.3.3 PAM instrumentation for each Function in Table T3.3.3.-1 shall be OPERABLE.

EO2c 1. Bases

a. The purpose of the PAM Instrumentation is to display information required by control room operators during and after accident conditions.

b. Provides control room operators with information to:

1) determine if safety systems are performing their safety function.

2) determine the potential for loss of radioactivity barriers.

3) initiate action necessary to protect the public.

2. Applicability

a. The PAM Instrumentation Functions are required as identified in Table T3.3.3.-1.

b. PAM monitoring instrumentation provides information for the ranges that may exist during the extreme conditions postulated to occur during and after some accidents.

3. Functions

a. Drywell Temperature

b. Suppression Chamber/Torus Air Temperature

c. Suppression Chamber/Torus Water Level (wide range)

d. Suppression Chamber/Torus Water Level (narrow range)

e. Suppression Chamber/Torus Pressure

f. Control Rod Position (Green full in lights)

g. Neutron Monitoring (3 cps - 100% power)

1) SRM (including recorder)
2) IRM (including recorders)
3) APRM (including recorders)

h. Elevated Release Point Monitor (high range noble gas)

i. Turbine Building Ventilation Exhaust Monitor (high range noble gas)

j. Radwaste/Augmented Radwaste Exhaust Monitor (high range noble gas)

k. Safety and Relief Valve Position Indication

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K1.07
	Importance Rating	3.1

Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) – K1.07 Leak detection

Question: 53

Following a plant transient the following conditions are present:

- Reactor Core Isolation Cooling (RCIC) is injecting at 400 gpm following an automatic initiation.
- A steam leak develops in the RCIC Room.
- RCIC Room area temperature is 195°F and rising.

What is the effect on RCIC?

A. Only RCIC-MO-16 (RCIC STEAM SUPPLY OUTBOARD ISOLATION VALVE) closes and the RCIC turbine trips.

B. Only RCIC-MO-15 (RCIC STEAM SUPPLY INBOARD ISOLATION VALVE) closes and the RCIC turbine coasts down (no trip).

C. RCIC-MO-15 and RCIC-MO-16 (RCIC STEAM SUPPLY INBOARD and OUTBOARD ISOLATION VALVE) close and the RCIC turbine trips.

D. RCIC-MO-15 and RCIC-MO-16 (RCIC STEAM SUPPLY INBOARD and OUTBOARD ISOLATION VALVE) close and the RCIC turbine coasts down (no trip).

Answer:

C. RCIC-MO-15 and RCIC-MO-16 (RCIC STEAM SUPPLY INBOARD and OUTBOARD ISOLATION VALVE) close and the RCIC turbine trips.

Explanation:

The following conditions will cause an automatic RCIC system isolation, also called a Group 5 isolation:

- RCIC steam supply low pressure (≥ 61 psig)
- RCIC steam supply line high flow (≤ 288% of rated + 6 sec. TD)
- RCIC steam line high space temperature (≤195°F)

When a Group 5 isolation occurs the RCIC Steam Supply Line Inboard and Outboard Isolation valves (MO-15 & 16) close, the RCIC turbine trips and the Minimum Flow valve (MO-27) closes.

Distracters:

A. This option is incorrect because RCIC-MO-15 also closes. The actions specified in this option are the actions that occur when a half group 5 isolation occurs on channel A. These are also the automatic actions that occur when the manual isolation button is depressed which only functions when an initiation is present. Since these actions are very specific and occur only when an initiation signal is present, as is the case here, a candidate may choose this option believing that with the initiation signal present the isolation is only on channel A.

B. This option is incorrect because RCIC-MO-16 also closes and the RCIC turbine trips. The actions specified in this option are those that occur with a half group 5 isolation on the B channel. A candidate may believe that with the initiation signal present the turbine coasts down due to the governor valve closing, rather than trip due to the trip throttle valve rapidly closing.

D. This option is incorrect because the RCIC turbine trips when the isolation signal is present. The isolation signal is a direct turbine trip signal. The candidate may believe the governor valve closes allowing the turbine to coast down rather than the trip throttle valve rapidly closing.

Technical Reference(s): SOP 2.2.67, Reactor Core Isolation Cooling System, Rev. 68

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Reactor Core Isolation Cooling/COR002-18-02 R25

10. Predict the consequences of the following on the RCIC system:

    n. Steam line break (Steam Tunnel/RCIC Room)

OPS Containment COR002-03-02 R30

6. Describe the interrelationship between PCIS and the following:

    f. RCIC

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 A3.08
	Importance Rating	3.0

201001 CRD Hydraulic – Ability to monitor automatic operations of the control rod drive hydraulic system including: (CFR: 41.7) A3.08 Drive water flow

Question: 54

The Plant is performing a Reactor Startup.

- Control Rod 26-27 is selected.
- A single notch withdrawal signal is applied to Control Rod 26-27.

What is the flow indication on Panel 9-5 CRD-FI-305, (**Drive Water Flow**) while the rod is **withdrawing**?

A. 2 gpm

B. 4 gpm

C. 6 gpm

D. 8 gpm

Answer:

A. 2 gpm

Explanation:

DRIVE WATER LINE: This line connects the drive water header to the HCU manifold. It is normally pressurized to 265 psi above reactor pressure.

WITHDRAWAL LINE: The withdrawal line connects the HCU manifold to the CRDM above piston area. The withdrawal line is pressurized with drive water when the associated CRDM is being withdrawn.

The insert solenoid valve has a throttle valve set to allow the normal insertion flow rate through that stabilizing valve, when it is open (energized). When there is no rod movement, the flow through the insert solenoid valve will be approximately 4 gpm. When there is an insert rod signal from the REACTOR MANUAL CONTROL SYSTEM (RMCS), the insert solenoid valve will close, balancing the 4 gpm flow directed to an HCU for normal insertion of a control rod.

The withdrawal solenoid valve has a throttle valve set to allow the normal withdrawal flow rate through the stabilizing valve when it is open (energized). When there is no rod movement, the flow through the withdrawal solenoid valve will be approximately 2 gpm. A withdraw signal from RMCS will close the valve balancing the 2 gpm flow directed to an HCU for normal withdrawal of a control rod.

Flow from the stabilizing valves passes through a local flow indicator. This local indication is used to adjust the throttle valves and verify proper operation. The normal reading, with no rod movement, is 6 gpm.

When the insert valve shuts, the withdrawal valve is still open; therefore the flow indicator will show 2 gpm flow through the stabilizing valves. When the withdrawal valve shuts, the insert valve is still open; therefore the flow indicator will show 4 gpm flow through the stabilizing valves.

Distracters:

B. This option is incorrect because on an insert the flow is 4 gpm. The candidate who confuses the expected insert flow with withdraw flow would choose this option.

C. This option is incorrect because 2 gpm is the expected flow during withdrawal. The candidate who confuses stabilizing flow (6 gpm) with that of withdrawal flow would choose this answer believing that when the rod withdraws that the entire stabilizing flow is routed to the withdrawal header.

D. This option is incorrect because only 2 gpm flow is required for withdrawal. A candidate may know that 6 gpm flow is the normal stabilizing flow and that when a withdrawal occurs that this flow is added to the 6 gpm (yielding 8 gpm) and choose this option.

Technical Reference(s): OPS Control Rod Drive Hydraulics/COR002-04-02, Rev. 23

Proposed references to be provided to applicants during examination: none

Learning Objective: LO-9 Given a CRDH system component manipulation, predict and explain the changes in the following parameters:

h. CRD drive water flow

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (6)

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	201006 A4.04
Importance Rating	3.3

201006 RWM - Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.04 Rod withdrawal error indication

Question: 55

The Plant is conducting a Reactor startup with the following conditions present:

- Group 3 Rod Movement Sheet Insert Limit: 04
- Group 3 Rod Movement Sheet Withdraw Limit: 08
- Control Rod 26-27 is selected and is the first of four rods in the Group.
- When Control Rod 26-27 is withdrawn from 06 to 08, it double notches.

Which RWM IDT targets turn RED?

A. WITHDRAW BLOCK **only**.

B. WITHDRAW BLOCK and OUT OF SEQUENCE **only**.

C. WITHDRAW BLOCK, OUT OF SEQUENCE and SELECT ERROR **only**.

D. WITHDRAW BLOCK, INSERT BLOCK, OUT OF SEQUENCE and SELECT ERROR **only**.

Answer:

B. WITHDRAW BLOCK and OUT OF SEQUENCE **only**.

Explanation:

Rod 26-27 has moved beyond the Withdraw Limit of Group 3 (08) and is at position 10.

The RWM allows only 1 withdrawal error and will impose rod blocks to prevent further progress until the withdraw error is corrected.

<u>IDT Display Console</u>

The purpose of the IDT Display Console is to provide the necessary displays and controls which, in conjunction with the Operator's Panel, allow the operator to monitor and control RWM operation.

1.Insert Error Display Windows

Displays the coordinates, in xx-yy format, of the rods responsible for causing insert errors, if they exist.

2.Withdraw Error Display Window

Displays the coordinates, in xx-yy format, of the rod responsible for causing a withdraw error, if one exists.

3.Group Display Window

Displays the current latched group number, or 999 if rods have been withdrawn beyond the end-of-sequence.

4.Rod Display Window

Displays the coordinates, in xx-yy format, of the current selected rod.

5.Position Display Window

Displays the notch position of the current selected rod.

6.Out-of-Sequence Indicator

This indicator warns the operator of out-of-sequence (i.e., withdraw error) conditions between all-rods-in and end-of-sequence.

It does this by changing from a green indicator reading IN SEQUENCE to a red indicator reading OUT OF SEQUENCE if one or more withdraw errors exist.

7.Select Error Indicator

This indicator warns the operator that a rod is selected that either is not in agreement with the rod sequence or does not correct an existing error. (Refer to earlier definition for details.) The indicator changes from green to red when a select error is made between all-rods-in and end-of-sequence.

8.Insert Block Indicator

This indicator changes from green to red when an insert block is applied by the RWM program.

9.Withdraw Block Indicator

This indicator changes from green to red when a withdraw block is applied by the RWM program.

<u>RWM Group</u> - RWM groups are sequential subdivisions of the operating sequence. An RWM group consists of one or more control rods with a set of insert and withdraw position limits that apply to each rod in the group.

Note that the first two rod groups involve control rod withdrawal from 00 to 48 for all the rods in that group (typically 16 to 18 rods). After the first two rod groups are withdrawn to 48, the next two rod groups are withdrawn to 48 in four distinct steps (a.k.a., RWM Groups), starting with 00 to 04, followed by 04 to 08, then 08 to 12, and 12 to 48.

<u>Withdraw Error</u> - A withdraw error occurs when a rod contained in current latched group or any lower group is withdrawn past withdraw limit for the group **<u>or</u>** if a rod contained in a higher group is withdrawn past insert limit for the higher group.

RWM IDT displays one rod with a withdraw error in WITHDRAW ERROR window.

Even if more than one withdraw error exists in actuality, only one will be recognized and displayed on the RWM IDT.

The RWM will permit the operator to correct the displayed withdraw error ***only***, even when more than one withdraw error conditions are actually present. You *<u>cannot</u>* select and insert just any withdraw error you wish to choose; you can *<u>only</u>* insert the displayed withdraw error rod.

Whenever the RWM is OPERATING < LPSP, the existence of a WITHDRAW ERROR causes OUT OF SEQUENCE (red) and WITHDRAW BLOCK (red) to display.

If, under these same conditions, the selection of any other control rod (other than the one causing the WITHDRAW ERROR) will cause OUT OF SEQUENCE (red), SELECT ERROR (red), INSERT BLOCK (red) and WITHDRAW BLOCK (red) to display.

Whenever a WITHDRAW ERROR occurs in the mode of enforcement (OPERATING < LPSP), the RWM will ONLY permit error rods to be repositioned.

Distracters:

A. This option is incorrect because Control Rod 26-27 is beyond the allowable withdrawal position, therefore, RWM IDT Withdraw Block AND Out of Sequence targets turn red. The candidate may choose this option if they do not remember the In Sequence target also turns red.

C. This option is incorrect because with Control Rod 26-27 beyond the allowable withdrawal position only RWM IDT Withdraw Block AND Out of Sequence targets are red. The candidate may choose this if they believe the Select Error turns red also because control rod 26-27 is selected.

D. This option is incorrect because Control Rod 26-27 is beyond the allowable withdrawal position, therefore, only RWM IDT Withdraw Block AND Out of Sequence targets will turn red. The candidate may choose this if they believe the Select Error is Red due to the rod remaining selected and the insert error is present due to the rod double notching.

Technical Reference(s): OPS Rod Worth Minimizer COR002-26-02, Rev. 22

Proposed references to be provided to applicants during examination: none

Learning Objective:
COR002-26-02, OPS Rod Worth Minimizer

LO-01 State the purpose of the following items related to the Rod Worth Minimizer:

g. IDT Display Console

LO-05 Briefly describe the following concepts as they apply to the RWM:

g. Withdraw error

Question Source:	Bank #
	Modified Bank #
	New X
Question History:	Last NRC Exam n\a
Question Cognitive Level:	Memory or Fundamental Knowledge
	Comprehension or Analysis X
10 CFR Part 55 Content:	55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	2.2.36
	Importance Rating	3.1

202001 Recirculation
G2.2.36- Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
(CFR: 41.10 / 43.2 / 45.13)

Question: 56

Reactor Recirculation Pump B is secured for maintenance in Mode 1.

What Technical Specification LCO is required to be entered?

A. LCO 3.4.2 (Jet Pumps)

B. LCO 3.1.2 (Reactivity Anomalies)

C. LCO 3.4.1 (Recirculation Loops Operating)

D. LCO 3.2.1 (Average Planar Linear Heat Generation Rate)

Answer:

C. LCO 3.4.1 (Recirculation Loops Operating)

Explanation:
Placing the plant in single loop in Mode 1 requires entry into TS LCO 3.4.1.

Distracters:

A. This answer is incorrect due to the Jet pump LCO not required to be entered. This choice is plausible due to the candidate possibly confusing Jet pump loss of drive flow with operability. Additionally, surveillance monitoring is performed every 24 hours on the Jet pumps but the surveillance is only required for each operating recirculation loop. The candidate who confuses Jet Pump operability with RR operability would choose this answer.

B. This answer is incorrect due to Reactivity Anomalies LCO not required to be entered. This choice is plausible due to RR being a reactivity control system and single loop being an abnormal condition. The candidate who confuses RR being part of the Reactivity Anomalies LCO would choose this answer.

D. This answer is incorrect due to APLHGR LCO not required to be entered. This choice is plausible due to the RR LCO requiring LHGR & MCPR thermal limits single loop limits to be incorporated within 24 hours which is easily confused with APLHGR LCO. The candidate who confuses thermal limit TS would choose this answer.

Technical Reference(s):
Technical Specifications
2.2.68.1 (Reactor Recirculation System Operations), Rev. 76

Proposed references to be provided to applicants during examination: None

Learning Objective: Reactor Recirculation / COR002-22-02 R32

LO-2 Given conditions and/or parameters associated with the Reactor Recirculation system or the Recirculation Flow Control system, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operation are met.

OPS Tech Specs 3.4, Reactor Coolant System (RCS)/INT007-05-05 R14

1. Given a set of plant conditions, recognize noncompliance with a section 3.4 LCO.

Question Source: Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

DIF 2

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	204000 K1.03
	Importance Rating	3.1

204000 RWCU - Knowledge of the physical connections and/or cause-effect relationships between reactor water cleanup system and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) - K1.03 Reactor feedwater system

Question: 57

Where does the Reactor Water Cleanup System (RWCU) return water to the reactor?

A. Via the suction side of A reactor recirculation pump.

B. Via the suction side of B reactor recirculation pump.

C. Via the A feedwater line downstream of the outboard check valve.

D. Via the B feedwater line downstream of the outboard check valve.

Answer:

C. Via the A feedwater line downstream of the outboard check valve.

Explanation:

When the return isolation valve is in its normal (open) position it allows return flow from the RHX to the Reactor Feedwater piping, where the water is returned to the reactor vessel via feedwater line "A".

Distractors:

A. This option is incorrect because RWCU returns water to the A FW line. RWCU does however get its water supply form the A RR loop so a candidate who only knows that there is a physical tie to the RR loop but not the purpose of that connection would choose this option.

B. This option is incorrect because the RWCU water returns to the reactor via the A FW line. The RWCU system does interconnect to the RR system so a candidate who only knows there is a connection but neither specific location nor the purpose of that connection would choose this option.

D: This option is incorrect because the RWCU system return water to the reactor is via the A FW line not the B FW line. A candidate who knows that the return is via the FW system but not which line would choose this option.

Technical Reference(s):	Procedure 2.2.66, Reactor Water Cleanup System, Rev. 104

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Reactor Water Cleanup COR001-20-01

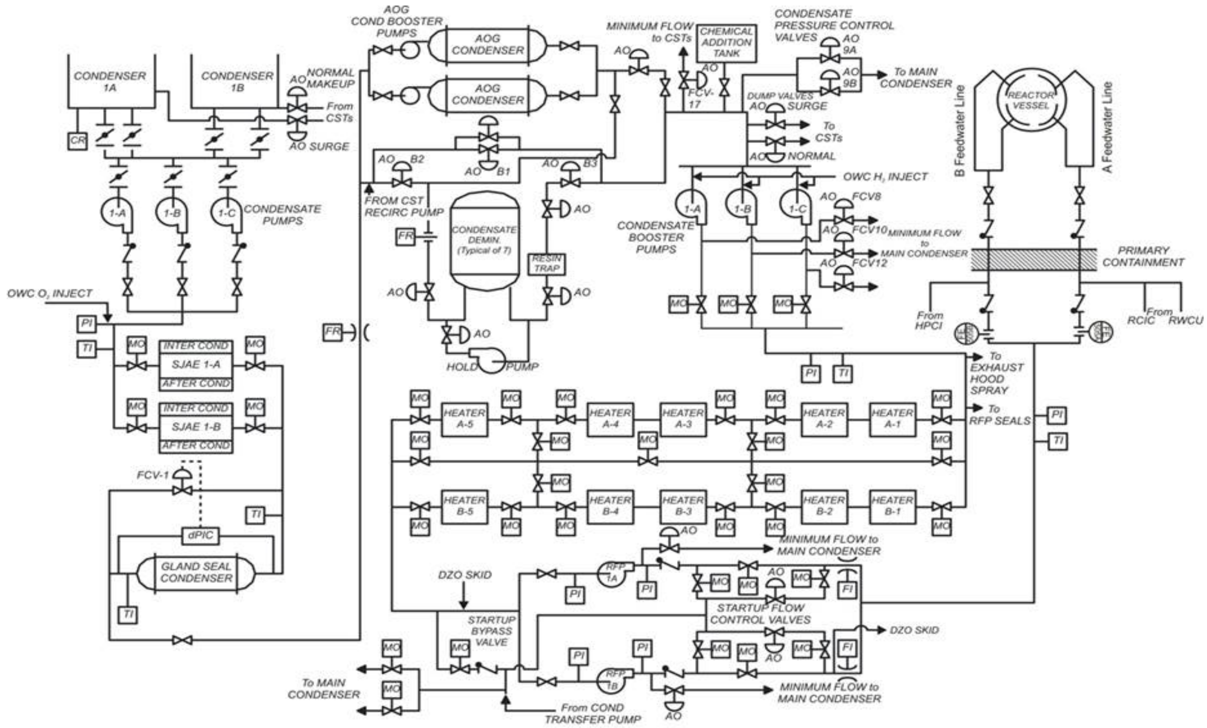
LO-4 Briefly describe the interrelationship between the RWCU system and the following:

j. Reactor Feedwater system

Question Source:	Bank # Modified Bank # New X
Question History:	Last NRC Exam n\a
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	55.41 (4)

Comments:

LOD 2



CONDENSATE and FEEDWATER COMPOSITE DIAGRAM

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 K2.01
	Importance Rating	2.8

216000 Nuclear Boiler Inst. - Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Analog trip system: Plant-Specific

Question: 58

What supplies power to the switch contact logic of NBI-LIS-57A (PCIS initiation logic)?

A. AA-1

B. NBPP A

C. RPSPP1A

D. 24V Power Panel DC-A

Answer:

C. RPSPP1A

Explanation:

The Wide Range level instrument NBI-LIS-57A is a Yarway type level indicating switch located on Local Rack LRP-PNL-(25-5A) in the Reactor Building. This switch is powered from 120 VAC RPSPP1A. This switch provides partial initiation to the following functions: Alternate Rod Insertion/Recirculation Pump Trip (ARI/RPT), Groups 1, 3, 6 and 7 isolations as well as Control Room Emergency Filtration System (CREFS) initiation.

Distracters:

A. This answer is incorrect because AA-1 does not supply power to any NBI level instrument or the logic power to any NBI level switch contact. This answer is plausible because panel AA-2 supplies power to narrow range level instruments and the candidate may confuse these two panels and choose this answer.

B. This answer is incorrect because NBPP-1A supplies RPV level Indicators NBI-LI-94A and NBI-LR/PR-97. This power supply is in the same division of power as NBI-LIS-57A, and is therefore plausible.

D: This answer is incorrect because the 24 VDC A supplies neutron monitoring and radiation monitor trip auxiliaries. This answer is plausible because it powers instruments on the same panel and close proximity to the RPV level instruments.

Technical Reference(s): OPS Nuclear Boiler Instrumentation/COR002-15-02 Rev. 26
SOP 2.2.22 Rev. 70 Vital Instrument Power System

Proposed references to be provided to applicants during examination: none

Learning Objective:

COR002-15-02 Rev. 26, OPS Nuclear Boiler Instrumentation

LO-05 Predict the consequences of the following on the NBI:

k. Loss of AC power

Question Source:	Bank #	
	Modified Bank #1145 (See attached)	
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (7)	

Comments:

LOD 4

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	1145	00	07/12/1999	06/15/2005	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	4	Multiple Choice	

Topic Area	Description
Systems	NBI, power supplies

Related Lessons
COR0021502 OPS NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001050K	Predict the consequences of the following on the NBI: Loss of AC power

Related References
NONE

Related Skills (K/A)	
216000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) Analog trip system: Plant-Specific (2.8/2.8)

QUESTION: 16 1145 (1 point(s))

The plant is operating at full power when indication is lost (failed downscale) on the following level instruments in the Main Control Room:

- Fuel Zone level indicators	LI-91A, C
- Wide Range level indicators	LI-85A, C
- Steam Nozzle Range indicator	LI-92

This is an indication of a loss of power on . . .

a. Critical Power Panel (CPP).

b. No Break Power Panel (NBPP).

c. Critical Control Panel 1A (CCP-1A).

d. Critical Control Panel 1B (CCP-1B).

ANSWER: 16 1145

c. Critical Control Panel 1A (CCP-1A).

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 K3.02
	Importance Rating	3.5

226001 RHR/LPCI: CTMT Spray Mode - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: containment spray system mode will have on following: (CFR: 41.7 / 45.4) - K3.02 Containment/drywell/suppression chamber temperature

Question: 59

The Plant is operating at power when a steam leak occurs in the Drywell.

- A reactor scram occurs.
- RHR is spraying the torus and drywell.
- Torus pressure is 11 psig lowering 1 psig/minute.
- Drywell temperature is 245°F lowering slowly.

A logic failure has caused the loss of spray valve control permissive.

What is the effect on drywell temperature?

Drywell temperature …

A. rises due to the loss of Torus Spray.

B. rises due to the loss of Drywell Spray.

C. lowers due to Torus Spray valves going full open.

D. lowers due to Drywell Spray valves going full open.

Answer:

B. rises due to the loss of Drywell Spray.

Explanation:
RHR must have spray valve control in order to spray containment. With a loss of the permissive, the spray valves close and the loss of containment cooling occurs. Drywell temperature rises due to the steam leak continuing.

A. This answer is incorrect because the loss of torus spray has little effect on drywell temperature. With the pressure suppression function remaining intact, spraying the torus air space has little effect on drywell temperature. This answer is plausible if there were a loss of pressure suppression function, then drywell temperature would rise. The candidate who does not understand the pressure suppression function would select this answer.

C. This answer is incorrect because the Torus Spray valves receive a closure signal on the loss of spray valve permissive. This answer is plausible because some RHR valves receive an open signal with a LPCI signal present. The candidate who does not know the effects of loss of spray valve control would select this answer.

D. This answer is incorrect because the Drywell Spray valves receive a closure signal on the loss of spray valve permissive. This answer is plausible because some RHR valves receive an open signal with a LPCI signal present. The candidate who does not know the effects of loss of spray valve control would select this answer.

Technical Reference(s): OPS Residual Heat Removal System/COR002-23-02
EOP-5.8.7 Rev.29 Primary Containment Flooding/Spray Systems
SOP 2.2.69.3 Rev.46
EOP-5.8 Attachment 2 EOP/SAG Graphs Rev.15

Proposed references to be provided to applicants during examination: None

Learning Objective:

Per COR002-23-02, OPS Residual Heat Removal System
7 Given a specific RHR system malfunction, determine the effect on any of the following:

d. Drywell parameters (pressure, temperature)

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments: LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 K4.05
	Importance Rating	2.8

230000 RHR/LPCI: Torus/Pool Spray Mode - Knowledge of RHR/LPCI: torus/suppression pool spray mode design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) - K4.05 Pump minimum flow protection

Question: 60

The plant is operating at 100% power when a small break LOCA occurs. The following conditions are present:

- RHR Loop A is in Torus Spray and Suppression Pool Cooling.
- Torus pressure is 2.3 psig and lowering slowly.

As flow on RHR loop A is being lowered, what condition FIRST causes the RHR-MO-16A (LOOP A MIN FLOW BYP VLV) to open?

RHR Loop A flow is…

A. <2107 gpm with no time delay.

B. <2107 gpm for 3.5 seconds.

C. <490 gpm with no time delay.

D. <490 gpm for 3.5 seconds.

Answer:

B. <2107 gpm for 3.5 seconds.

Explanation:

Minimum Flow Valves RHR-MO-16A/B

The pump minimum flow valves, one in each loop, provide the necessary flow through the pump in order to prevent pump overheating. The RHR pump minimum flow control valves are normally open. The RHR-MO-16A valve closes when flow is ≥2107 gpm for 3.5 seconds in the associated loop if RHR-MO-20 (RHR Cross-tie) is not full open, or ≥2107 gpm in <u>either</u> Loop A or B for 3.5 seconds if RHR-MO-20 is full open. The valve opens when flow is less than 2107 for 3.5 seconds on system shutdown.

Distracters:

A. This answer is incorrect because the minimum flow control valve opens when flow is less than 2107 gpm for 3.5 seconds in the associated loop. The valve responds after a time delay. If the candidate does not remember the time delay, then this answer may be selected.

C. This answer is incorrect because the RHR minimum flow control valve opens when flow is less than 2107 gpm for 3.5 seconds in the associated loop. The flow value of 490 is plausible because it is the HPCI system minimum flow setting. The no time delay is also associated with the HPCI system minimum flow valve. If the candidate confuses the HPCI system minimum flow value with the RHR minimum flow value, then this answer may be selected.

D. This answer is incorrect because the RHR minimum flow control valve opens when flow is less than 2107 gpm for 3.5 seconds in the associated loop. The flow value of 490 is plausible because it is the HPCI system minimum flow setting. The time delay of 3.5 seconds was used because it is the setting for the RHR minimum flow time delay. If the candidate confuses the HPCI system minimum flow value with the RHR minimum flow value, then this answer may be selected.

Technical Reference(s): SOP 2.2.69.3 RHR Suppression Pool Cooling and Containment Spray, Rev. 46

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Residual Heat Removal System/COR002-23-02

LO-3 Describe RHR system design feature(s) and/or interlocks which provide for the following:

c. Pump minimum flow protection

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K5.04
	Importance Rating	3.3

241000 Reactor/Turbine Pressure Regulator – Knowledge of the operational Implications of the following concepts as they apply to reactor/turbine pressure regulating system: (CFR: 41.5) K5.04 Turbine inlet pressure vs. reactor pressure.

Question: 61

During a startup the turbine is synchronized to the grid and the following conditions are present:

- Equalizing header pressure is 935 psig.
- Reactor Pressure is 940 psig.
- Reactor power is 25%.

As power is raised from 25% to 100%, how does the pressure at the equalizing header and the reactor change, if at all, during the power ascension?

Equalizing header pressure …

A. remains at 935 psig and reactor pressure rises to about 958 psig.

B. remains at 935 psig and reactor pressure rises to about 990 psig.

C. rises to about 958 psig and reactor pressure rises to about 958 psig.

D. rises to about 958 psig and reactor pressure rises to about 990 psig.

Answer:

D. rises to about 958 psig and reactor pressure rises to about 990 psig.

When the main turbine is on-line, Throttle Pressure is controlled by modulating the governor valves. The pressure target is set via the Human Machine Interface (HMI). The DEH system gain is configured in the controller calculations is set to 3.33%. This gain results in a 3.33% flow demand change for every 1 psi of error sensed at the main steam equalizing header. A 30 psi error change results in a 100% flow demand change. This feature makes the response of the Throttle Pressure linear. RPV Steam Dome pressure varies based on the response of the DEH system to power and is a function of steam flow head loss. Since the head loss is a function of volume flowrate squared, the rise in reactor pressure is greater that the rise in the equalizing header.

Distractors:

A. This option is incorrect because the equalizing header pressure rises as power is raised. Even though pressure setpoint remains the same the offset between pressure setpoint and the equalizing header pressure provides the error to drive the governor valves to the new position. Reactor pressure does rise as indicated but rises more than to 958 psig. A candidate may very well believe that since pressure setpoint is constant that equalizing header is constant and if that same candidate understands that reactor pressure rises due to head loss they would choose this option.

B. This option is incorrect because even though reactor pressure does rise to about 990 psig, the equalizing header pressure does not remain at 935 psig as power is raised. Even though pressure setpoint remains the same the offset between pressure setpoint and the equalizing header pressure provides the error to drive the governor valves to the new position so the offset between the setpoint and the actual equalizing header gets larger as power is raised. A candidate who understands the true response of reactor pressure but who does not understand that it is the offset between the pressure setpoint and equalizing header that provides the signal to open the governor valves would choose this option.

C. This option is incorrect because reactor pressure rises to greater than the new equalizing header pressure. The candidate who understands that equalizing header pressure rises but then fails to consider that there is head loss between the reactor and the equalizing header would choose this option.

Technical Reference(s): SOP 2.2.77.1 Rev.35, Digital Electro-Hydraulic Control System
SOP 2.2.77 Rev.111, Turbine Generator

Proposed references to be provided to applicants during examination: None

Learning Objective: COR002-09-02 Rev.17

5. Explain the interrelation between the following parameter sets, and describe how their interrelationship affects operation of the DEH Control system.

   b. Turbine inlet pressure vs. reactor pressure

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 (5)

Comments:
LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 Reactor Feedwater K6.04
	Importance Rating	2.8

259001 Reactor Feedwater - Knowledge of the effect that a loss or malfunction of the following will have on the reactor feedwater system: (CFR: 41.7 / 45.7) - K6.04 Extraction steam

Question: 62

The plant is operating at 95% power on the 100% rod line when the Non-Return Isolation Check Valve to Feedwater Heater 1-A-5 goes full closed due to a short in its motor operator.

When conditions stabilize, how have feed water temperature and the operating rod line changed?

	Feedwater temperature	Operating Rod Line
A.	rises	< 100%
B.	lowers	< 100%
C.	rises	> 100%
D.	lowers	> 100%

Answer:

D.	lowers	> 100%

Explanation:

The Extraction Steam system conducts steam from Main Turbine connections to two parallel Feedwater heater strings to improve the overall efficiency of the reactor by preheating the incoming feedwater and reducing the reactor heat load. The NON-RETURN ISOLATION CHECK VALVES (**NRV**s) are installed in the extraction steam supply lines to the Feedwater Heaters. NRV-1 is designed with an integral motor operator and when closed will essentially function as a stop check valve. When NRV-1 goes closed, feedwater Heater 1-A-5 heat transfer rate significantly lowers. This causes the FW temperature to lower.

ESAR Vol. V., Section XIV, Part 5.2.1; Loss of Feedwater Heating

A decrease in feedwater temperature due to loss of feedwater heating would result in a core power increase. This power rise (at the same reactor recirculation flow) would raise the operating rod line. So at a constant flow, power is higher and therefore the operating rod line is higher.

Distractors:

A. This answer is incorrect because with the loss of extraction steam, feedwater is not being preheated so feedwater temperature lowers. And the operating rod line rises due to the now higher reactor power. A candidate who believes that the malfunctioning NRV causes more steam to be admitted to the heater (as is the case with closure of a heater steam dump valve) would choose this answer.

B. This answer is incorrect because the operating rod line is higher not lower after conditions stabilize. The candidate who understands the effect of the NRV malfunction but not the eventual effect on the reactor may choose this answer.

C. This option is incorrect because the feed water temperature lowers. The candidate who does not understand the extraction steam system and evaluates the effect of the NRV as that of a dump valve would choose this answer.

Technical Reference(s): 2.4EX-STM, Rev. 18
2.1.10 Station Power Changes, Attachment 1 Power to Flow Map, Rev. 107

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Extraction Steam and Heater Drains COR001-04-01

LO-06 Given a specific extraction steam and heater drains malfunction, determine the effect on any of the following:

d. Reactor Feedwater

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 A1.08
	Importance Rating	3.8

239001 Main and Reheat Steam System: Ability to predict and/or monitor changes in parameters associated with operating the MAIN AND REHEAT STEAM SYSTEM controls including: A1.08 Reactor pressure (CFR: 41.5 / 45.5)

Question: 63

Reactor power is reduced to support performance of Surveillance Procedure 6.MS.201 (Main Steam Isolation Valve Operability Test).

The following conditions exist when the surveillance is started:
Reactor power is 65%.
Reactor pressure is 980 psig.

Test Pushbutton is depressed and sticks in the depressed position.

How long does it take the MSIV to close from the time the pushbutton is first depressed and what is the final reactor pressure when conditions stabilize?

	Time to Close	Final Reactor Pressure
A.	< 5 seconds	>980 psig
B.	< 5 seconds	<980 psig
C.	> 20 seconds	>980 psig
D.	> 20 seconds	<980 psig.

Answer:
C. > 20 seconds >980 psig
.

Explanation: Requires operating a Main Steam Isolation Valve and predicting the reactor pressure change.

MSIV Spring Only Closure Test vents air off the MSIV operator to allow closure by spring pressure only which takes greater than 20 seconds to reach full closure. With the pushbutton in the depressed position and when conditions stabilize with the MSIV closed, reactor pressure will be higher due to the increased head loss of 3 Main Steam Lines vs. 4 open Main Steam Lines.

Distracters:

A. This answer is incorrect due to MSIV slow closure being greater than 5 seconds. This choice is plausible if confused with TS maximum closure time of 5 seconds. The candidate who confuses MSIV slow closure time and correctly identifies reactor pressure response would choose this answer.

B. This answer is incorrect due to MSIV slow closure being greater than 5 seconds and reactor pressure not returning to the original pre-closure pressure. This choice is plausible if confused with TS maximum closure time of 5 and if confused with equalizing header pressure which would be maintained at the original pressure by the DEH system. The candidate who confuses MSIV slow closure time and confuses equalizing header pressure or DEH response would choose this answer.

D. This answer is incorrect due to reactor pressure not returning to the original pre-closure pressure. This choice is plausible if confused with equalizing header pressure which would be maintained at the original pressure by the DEH system. The candidate who correctly identifies MSIV slow closure time and confuses equalizing header pressure or DEH response would choose this answer.

Technical Reference(s):
Procedure 6.MS.201 (Main Steam Isolation Valve Operability Test), Rev. 15
Procedure 2.2.56 (Main Steam System), Rev. 49

Proposed references to be provided to applicants during examination: None

Learning Objective: COR0021402001070F - Predict the consequences of the following items on the MAIN STEAM SYSTEM: Closure of one or more MSIV's at power

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (5)

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 A2.15
	Importance Rating	2.5

272000 Radiation Monitoring – Ability to (a) predict the impacts of the following on the radiation monitoring system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5) A2.15 Maintenance operations

Question: 64

Maintenance is being performed on RMP-RM-150B (OFFGAS RAD MONITOR B) while operating at rated power with the following annunciator in alarm:

OFFGAS RAD MON DOWNSCALE OR INOP	PANEL/WINDOW: 9-4-1/D-4

(1) Which of the following identifies the impact on the Off-Gas system if radiation levels rise causing RMP-RM-150A (OFF GAS RAD MONITOR A) to reach the Hi Hi Trip Setpoint?

(2) What action is required following the Off-Gas System isolation IAW Procedure 2.4OG (Off-Gas Abnormal)?

A. (1) Off-Gas isolates IMMEDIATELY.
(2) SCRAM and enter Procedure 2.1.5 (Reactor Scram).

B. (1) Off-Gas isolates IMMEDIATELY.
(2) Lower reactor power per Procedure 2.1.10 (Station Power Changes).

C. (1) Off-Gas isolates after a 15 minute time delay.
(2) SCRAM and enter Procedure 2.1.5 (Reactor Scram).

D. (1) Off-Gas isolates after a 15 minute time delay.
(2) Lower reactor power per Procedure 2.1.10 (Station Power Changes).

Answer:
C. (1) Off-Gas isolates after a 15 minute time delay.
(2) SCRAM and enter Procedure 2.1.5 (Reactor Scram).

Explanation:

Requires knowledge of maintenance activity impact on the OG system and Scram actions of 2.4OG. The off-gas stream is monitored by two radiation monitors: RMP-RM-150A and RMP-RM-150B. With one channel inoperable (mode switch not in operate or high voltage low) an alarm sounds (9-4-1/D-4) and the other channel downscale, HI-HI, or Inop starts a 15 minute timer. After 15 minutes, off-gas isolates. OG-254 closes after 15 minutes which isolates the off-gas system. IAW 2.4OG – if the OG system isolates due to Hi Radiation, Scram and enter Procedure 2.1.5.

Distracters:

A. This answer is incorrect due Off-gas does NOT isolate immediately under these conditions. This choice is plausible due to not recalling OG having a 15 min time delay. The candidate that confuses OG isolation time delay and correctly identifies a scram is required following isolation due to valid radiation levels would choose this answer.

B. This answer is incorrect due Off-gas does NOT isolate immediately under these conditions and power reduction not required under the given conditions. This choice is plausible due to not recalling OG having a 15 min time and power reduction being required to maintain main condenser vacuum and to reduce the OG radiation levels prior to OG isolation. The candidate who confuses OG isolation time frame and does not recognize 2.4OG Scram action would choose this answer.

D. This answer is incorrect because reducing power not required under the given conditions. This choice is plausible due to power reduction being required to maintain main condenser vacuum and to reduce the OG radiation levels prior to OG isolation. The candidate who correctly identifies OG isolation time delay and does not recognize 2.4OG Scram action would choose this answer.

Technical Reference(s):
Procedure 2.4OG (Off-Gas Abnormal), Rev. 22

Proposed references to be provided to applicants during examination: None

Learning Objective: COR001-18-02

7. Given a specific Radiation Monitoring system malfunction, determine the effect on any of the following:
   b. Station Area Radiation monitoring
11. Predict the consequences of the following items on the Radiation Monitoring system.
   g. Maintenance/Surveillance operations

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (5)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	288000 A3.01
	Importance Rating	3.8

288000 Plant Ventilation - Ability to monitor automatic operations of the plant ventilation systems including: (CFR: 41.7 / 45.7) - A3.01 Isolation/initiation signals

Question: 65

While operating at power, multiple smoke detectors actuate in the SW Pump Room near the ceiling and in the fan exhaust duct.

How does the SW Pump Room HVAC respond?

A. The supply fans trip and the isolation dampers automatically open.

B. The supply fans trip and the isolation dampers automatically close.

C. The supply fans continue to run and the isolation dampers automatically open.

D. The supply fans continue to run and the isolation dampers automatically close.

Answer:

B. The supply fans trip and the isolation dampers automatically close.

Explanation:

A smoke detector (SD-1003A & B) in the exhaust duct of each unit can detect smoke or Halon in its respective exhaust duct. Actuation of one or both of these detectors will shut down both supply fans. Upon actuation of a smoke, flame, or thermal detector located on the SW Pump Room ceiling, the Halon System will enter the alarm mode. If a second detector actuates, the Halon System enters the predischarge mode, the isolation dampers in the system close to stop ventilation flow into and out of the room.

Distracters:

A. This answer is incorrect because the supply fans trip and the isolation dampers close. The candidate may believe that this is plausible if he/she believes that the isolation dampers automatically open to help naturally remove smoke. The only time that the dampers automatically open is following reset of the initiating conditions.

C. This answer is incorrect because the supply fans trip and the isolation dampers close. The candidate may believe that this is plausible if he/she believes that the supply fans continue to run with the dampers automatically open to remove smoke from the area. The only time that the dampers automatically open is following reset of the initiating conditions.

D. This answer is incorrect because the supply fans trip and the isolation dampers close. The candidate may believe that this is plausible if he/she believes that the supply fans continue to run with the dampers closed to direct fresh air into the SW Pump room.

Technical Reference(s): Alarm Procedure 2.3_FP-2 FP-2/F6 Rev. 5

Proposed references to be provided to applicants during examination: none

Learning Objective: OPS Heating, Ventilation and Air Conditioning COR001-08-01 R28

6. Describe the interrelationships between HVAC systems and the following:
   j. Fire Protection

11. Describe the HVAC design features and interlocks that provide for the following:
   d. System damper alignment

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments:

LOD 3

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Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.8
	Importance Rating	3.4

Ability to coordinate personnel activities outside the control room. (CFR: 41.10 / 45.5 / 45.12 / 45.13)

Question: 66

With the plant at power, a manual valve is closed maintaining Primary Containment IAW TS LCO 3.6.1.3 (PCIVs).

How is this valve controlled IAW Procedure 2.0.1 (Plant Operations Policy)?

The Reactor Operator directs the Plant Operator to…

A. open the valve and remain at the valve ONLY.

B. open the valve and remain at the valve maintaining continuous communication with the Control Room ONLY.

C. open the valve, remain at the valve maintaining continuous communication with the Control Room, and instructed to close valve in event of an accident condition.

D. open the valve, remain at the valve maintaining continuous communication with the Work Control Center, and instructed to close valve in event of an accident condition.

Answer:
C. open the valve, remain at the valve maintaining continuous communication with the Control Room, and instructed to close valve in event of an accident condition ONLY.

Explanation:
This question requires the Reactor Operator to coordinate personnel activities outside the control room (Plant Operator local valve operations under TS Administrative Controls).
Procedure 2.0.1 (Plant Operations Policy) provides the following guidance:
Isolation valves closed to satisfy LCO 3.6.1.3 may be re-opened on an intermittent basis following administrative controls:

- A person shall be stationed at valve controls while valve is open.
- If valve is being controlled outside of Control Room, person at valve controls shall be in continuous communication with Control Room.
- Person at valve controls shall be instructed to close valve in event of an accident condition.

Distracters:

A. This answer is incorrect due to not providing ALL the procedural requirements; specifically continuous communication and direction for valve operation during emergency conditions is not included. This choice is plausible if the candidate does not remember all of the requirements for opening a PCIV under administrative controls.

B. This answer is incorrect due to not providing ALL the procedural requirements; specifically direction for valve operation during emergency conditions is not included in the selection. This choice is plausible if the candidate does not remember all of the requirements for opening a PCIV under administrative controls.

D. This answer is incorrect because the Work Control Center is provided as an alternate selection to the Control Room. This choice is plausible if all the requirements for opening a PCIV under administrative controls are not specifically known the candidate thinks that the Work Control Center may control this administrative task.

Technical Reference(s):
Procedure 2.0.1 (Plant Operations Policy), Rev. 62

Proposed references to be provided to applicants during examination: None

Learning Objective:

SKL0080102 Ops Watchstanding Principles for Licensed Operators

0010400 Briefly describe the administrative controls for primary or secondary containment manual valve opening or associated cap removal when primary or secondary containment is required.

Question Source:	Bank # Modified Bank # New X	
Question History:	Last NRC Exam	n\a
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.18
	Importance Rating	3.6

Ability to make accurate, clear, and concise logs, records, status boards, and reports. (CFR: 41.10 / 45.12 / 45.13)

Question: 67

The Reactor has just been declared CRITICAL during plant startup. In addition to:

- Date/Time
- Control Rod Number
- Control Rod Position
- Sequence
- Reactor Pressure

What additional criticality information is required to be logged in the Control Room Operators Log IAW Procedure 2.1.1 (Startup Procedure)?

A. SRM Counts **AND** IRM Overlap.

B. SRM Counts **AND** Moderator Temperature.

C. Reactor Period **AND** IRM Overlap.

D. Reactor Period **AND** Moderator Temperature.

Answer:

D. Reactor Period **AND** Moderator Temperature.

Explanation:
This answer requires knowledge/ability of required accurate criticality log entries. Procedure 2.1.1 requires the following narrative log criticality data – Log control rod number, control rod position, moderator temperature, reactor pressure, sequence, reactor period, and time on Procedure 10.13, Attachment 1, and in Control Room Operator's log.

Distracters:

A. This answer is incorrect due to SRM Counts and IRM overlap not required to be logged in the narrative log for criticality data. This choice is plausible due to SRM & IRM chart recorders are required to be annotated and IRM overlap required to be document in Procedure 2.1.1. The candidate who confuses annotating charts and documenting IRM overlap would choose this answer.

B. This answer is incorrect due to SRM Counts not required to be logged in the narrative log for criticality data. This choice is plausible due to SRM & IRM chart recorders are required to be annotated IAW Procedure 2.1.1. The candidate who confuses annotating charts and correctly identifies Moderator Temperature would choose this answer.

C. This answer is incorrect because IRM overlap is not required to be logged in the narrative log for criticality data. This choice is plausible due to IRM overlap is required to be documented in Procedure 2.1.1 but not at the time of criticality. The candidate who correctly identifies Period and confuses documenting IRM overlap would choose this answer.

Technical Reference(s): GOP 2.1.1, Startup Procedure, Rev. 178

Proposed references to be provided to applicants during examination: none

Learning Objective:

INT0320104 CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures

00A0800 Describe the required actions to be completed upon achieving criticality as described in Procedure 2.1.1, Startup Procedure.

Question Source:	Bank # 2377 Modified Bank # New	
Question History:	Last NRC Exam	n\a
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 (10)	

Comments:

LOD 3

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.31
	Importance Rating	4.6

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)

Question: 68

Which of the following identifies the location and indication that confirms the reactor is in Hot Shutdown when the Mode Switch is in the Shutdown position during a reactor cooldown?

A. Reactor pressure indicates 15 psig on Panel 9-5.

B. Feedwater Temperature indicates 200°F on Panel A.

C. RHR HX Inlet Temperature indicates 200°F on Panel 9-3.

D. Vessel Head Adjacent to Flange temperature indicates 240°F on Panel 9-21.

Answer:
A. Reactor pressure indicates 15 psig on Panel 9-5.

Explanation:
Requires knowledge of indication location and indication which supports the plant being in Mode 3, IAW TS Definitions - Hot Shutdown (Mode 3) is determined by Mode Switch position (Shutdown) and Average Reactor Coolant Temperature (>212°F) with all reactor vessel head closure bolts fully tensioned. The temperature of saturated steam at 15 psig is 249.5°F.

Distracters:

B. This answer is incorrect due to FW Temperature indicating 200°F. This choice is plausible due to Panel A being the location for FW Temperature (correct location) and confusing FW Temperature as a valid indication of Average Reactor Coolant Temperature and temperature required for HOT shutdown confused with COLD shutdown of $\leq$ 212°F. The candidate who correctly identifies FW Temperature indication is located on Panel A and confuses Cold Shutdown with Hot Shutdown temperature requirements would choose this answer.

C. This answer is incorrect due to RHR Hx inlet Temperature indicating 200°F. This choice is plausible due to Hx inlet Temperature being located on Panel 9-3 (correct temperature) and RHR Hx inlet temperature being reactor coolant temperature if indicating 200°F. The candidate who correctly identifies RHR Hx inlet temperature but confuses Cold Shutdown with Hot Shutdown temperature requirements would choose this answer.

D. This answer is incorrect because RPV metal temperatures are not specified on T.S. Table 1.1-1 when determining Reactor Modes. This answer is plausible because of the heat transport mechanism and temperature gradients involved across the RPV shell wall. The candidate who incorrectly believes that RPV metal temperatures instead of average reactor coolant temperature would provide indication of Reactor Mode conditions would choose this answer.

Technical Reference(s):
TS Table 1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: INT00705010010400 From memory, given a set of plant conditions, determine the plant MODE.

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam n\a

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 3

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	> 212
4	Cold Shutdown[(a)]	Shutdown	≤ 212
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.13
	Importance Rating	4.1

Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)

Question: 69

Which of the following identifies the FIRST two tagging order steps for placing a system pump under clearance, and the reason for this order IAW Procedure 0.9, Tagout?

OPEN the pump breaker and then close the…

A. suction valve to minimize draining time.

B. discharge valve to minimize draining time.

C. suction valve to protect low pressure components.

D. discharge valve to protect low pressure components.

Answer:

D. discharge valve to protect low pressure components.

Explanation:
Requires knowledge of tagging procedure 0.9. Procedure 0.9 provides guidance for pump tagging to remove the power source first. If isolating the pump, the discharge valve is closed before the suction valve to prevent possible over-pressurization of low pressure components on the suction side.

Distracters:

A. This answer is incorrect due to the suction valve being closed prior closing the discharge valve and minimizing draining time. This choice is plausible due to confusing the reason for closing the discharge valve prior to the suction and minimizing draining time is desired but not the reason. The candidate who confuses valve closure order and reason for the order would choose this answer.

B. This answer is incorrect because minimizing draining time is not the reason for the order of operations. This choice is plausible due to confusing the reason for closing the discharge valve prior to the suction and minimizing draining time is desired but not the reason. The candidate who correctly identifies the valve closure order and confuses the reason for the order would choose this answer.

C. This answer is incorrect due to the suction valve being closed prior closing the discharge valve. This choice is plausible due to confusing the reason for closing the discharge valve prior to the suction. The candidate who confuses valve closure order and correctly identifies the reason for the order would choose this answer.

Technical Reference(s): Procedure 0.9, Tagout, Rev. 85

Proposed references to be provided to applicants during examination: None

Learning Objective: SKL00803020010600 Describe the proper sequence for hanging and picking up tags with regards to Tagging Orders.

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

LOD 3

Comments:

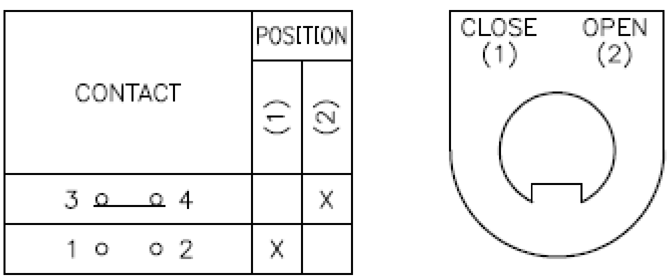
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.41
	Importance Rating	3.5

Ability to obtain and interpret station electrical and mechanical drawings.
(CFR: 41.10 / 45.12 / 45.13)

Question: 70

An Operator is preparing to write a Clearance Order on a Solenoid Valve.

Where can the Operator **(1)** obtain Controlled Copies of the electrical and mechanical drawings needed to prepare the Clearance and when interpreting the electrical drawing **(2)** what is the status of Contact 3-4 when the switch in the CLOSE position?



NAME PLATE "A"

	(1)	**(2)**
A.	Control Room	Open
B.	Operations Support Center	Open
C.	Control Room	Closed
D.	Operations Support Center	Closed

Answer:

A. Control Room Open

Explanation:

GENERATING A TAGOUT SECTION

Controlled Distribution drawings are located in the areas identified below:

a. Aperture cards located in Technical Support Center (TSC), Information Resource Center (IRC), AND Central Alarm Station (CAS).

b. Full size drawings with copies located in the TSC, EOF, Control Room, Simulator Control Room, I&C Shop, E-Shop, Work Control Center (WCC), and Planning.

The embedded switch development depicts a switch with two positions: Close (1) and Open (2). An “x” in the column is used to determine the status of the contacts (1-2, 3-4) associated with the switch.

In the given switch development matrix, an “x” is in column (2), which corresponds to the Open position. The contacts 3-4 will be in the condition as drawn (i.e. closed). Should the switch be in the Close position, column (1), it does not have an “x” in it; therefore contacts 3-4 would be open.

Distractors:

B. This option is incorrect because the Operations Support Center is not on the Controlled Drawing distribution list. Contact 3-4 is open when the switch is in the Close positon. A candidate may believe that because the operations support center uses many drawing that this would be where controlled copies exist would choose this option. It should also be noted that this may be confused with the Operational Support Center which is in the same area as the TSC and does maintain controlled documents for emergency purposes.

C. This option is incorrect because contact 3-4 is open when the switch is in the Close positon. A candidate may believe that the position of the x in the switch development means open and would choose this option. This answer is plausible because the drawing location is correct.

D. This option is incorrect because the Operations Support Center is not on the Controlled Drawing distribution list and contact 3-4 is open when the switch is in the Close position. A candidate may believe that the position of the x in the switch development means open and that the OSC is on the distribution list and choose this option. This answer is plausible because the switch contact 3-4 do exist. It should also be noted that this may be confused with the Operational Support Center which is in the same area as the TSC and does maintain controlled documents for emergency purposes.

Technical Reference(s): 3.DRAWING, Drawing Control, Rev. 4
0.9, Tagout, Rev. 85
OTH015-09-08, Electrical Print Reading for Clearance Order and
Circuit Evaluation Applications, Rev. 0

Proposed references to be provided to applicants during examination: none

Learning Objective: OTH015-09-08/Electrical Print Reading for Clearance Order and Circuit Evaluation Applications

5. Describe the basic types of electrical drawings.

   a. Determine the status of prints and actions required when generating a clearance order.

   b. Determine where electrical prints are available and how to retrieve them.

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Question: 71

What is the accumulated dose value above which a tour member is first required to have an NRC Form 5 or equivalent issued?

A. 100 mrem

B. 500 mrem

C. 1000 mrem

D. 5000 mrem

Answer:

A. 100 mrem

Explanation:

9.ALARA.1 Personnel Dosimetry and Occupational Radiation Exposure Program step 5.5 specifies the dose value which requires an NRC Form 5 or equivalent to be issued. The accumulated dose value specified is >100 mrem. 9.ALARA.13 provides the instructions for completing the NRC Form 5.

Distractors:

B. This option is incorrect because the selection is above the >100 mrem accumulated value. This choice is plausible because it is the accumulated dose allowed for a pregnant female per 10CFR20.1208. If the candidate confuses these two values this option would be chosen due to 500 mrem being a familiar dose value.

C. This option is incorrect because the selection is above the >100 mrem accumulated value. This choice is plausible because it is the CNS administrative dose limit for the year. If the candidate confuses these two values this option would be chosen due to 1000 mrem being a familiar dose value.

D. This option is incorrect because the selection is above the >100 mrem accumulated value. This choice is plausible because it is the annual allowed 10CFR20.1201 TEDE dose. If the candidate confuses these two values this option would be chosen due to its potential familiarity.

Technical Reference(s): 9.ALARA.1 Personnel Dosimetry and Occupational Radiation Exposure Program.

Proposed references to be provided to applicants during examination: none

Learning Objective: INT032-01-15R06 OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures.

D. Procedure 9.ALARA.1, Personnel Dosimetry and Occupational Radiation Exposure Program

g. Monitoring Tour Groups

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (12)

Comments:

LOD: 4

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

2.3.5 – Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question: 72

What is/are the **MINIMUM** personnel monitoring requirement(s) for exiting a contaminated area and dressing in street clothing?

A. Perform a hand and foot frisk.

B. Perform a whole body contamination monitor scan.

C. Perform a whole body frisk **then** a whole body contamination monitor scan.

D. Perform a hand and foot frisk **then** a whole body contamination monitor scan.

Answer:

D. Perform a hand and foot frisk **then** a whole body contamination monitor scan.

Explanation:

Per 9.EN-RP-100

6.2 CONTAMINATION CONTROL IN CONTAMINATION AREAS (CAs)

6.2.1 Personnel exiting a contamination area shall perform, as a minimum, a hand and foot frisk as soon as practical upon exiting.

6.2.2 When exiting a highly contaminated area or a discrete radioactive particle area, a whole body frisk is done as soon as possible.

6.2.3 Perform a whole body frisk using a whole body contamination monitor or a frisker before personnel don any clothing not worn in a contaminated area.

6.2.4 When exiting the RCA, all persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole body contamination monitor and a gamma portal monitor.

The candidate should recognize that contamination control requires a minimum of a hand and foot frisk to reduce the potential for spread of contamination. The requirement to exit the RCA via a whole body monitor is always a requirement.

A. This answer is incorrect because it does not include the whole body monitor. The candidate could choose this distractor if he/she does not equate the whole body monitor with contamination control. This answer is plausible because performing hand and foot frisk is required.

B. This answer is incorrect because it does not include hand and foot frisk. The candidate could choose this distractor if he/she does not equate hand and foot frisk with contamination control. This answer is plausible because using the whole body monitor is required.

C. This answer is incorrect because it does not properly show the allowance for a hand and foot frisk. The candidate could choose this distractor if he/she did not remember the minimum requirement of the procedure. This answer is plausible because using the whole body monitor is required.

Technical Reference(s): 9.EN-RP-100, Radiation Worker Expectations, Rev. 4

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:	Bank #	X
	Modified Bank #	
	New	
Question History:	Last NRC Exam	Perry 2009
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 (11)	

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.4

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.
(CFR: 41.12 / 43.4 / 45.9 / 45.10)

Question: 73

An Operator has to enter a room to close a valve to stop a small water leak.

- The affected work area general radiation level is 1500 mrem/hour.

What is the required entry permit?
Is continuous RP coverage required during the entry?

	Entry Permit	Continuous RP Coverage
A.	SWP	NOT Required
B.	SWP	Required
C.	RWP	Required
D.	RWP	NOT Required

Answer:

B. SWP Required

Explanation:

Operations personnel are the only authorized persons to manipulate valves that are not controlled by Clearances. See yellow highlighted NOTE below.

Locked High Radiation Area - An area accessible to individuals in which radiation levels from sources external to the body could result in an individual receiving a deep dose equivalent > 1 rem (10 mSv) in 1 hour at 30 cm (~ 12") from the radiation source or from any surface that the radiation penetrates.

HIGH RADIATION AREA (HRA) ACCESS CONTROL

High Radiation Area entry points require a barricade to prevent inadvertent access.

If the barricade for a HRA must be temporarily removed, then a RP Technician may maintain direct "line-of-sight" surveillance of the access to the HRA until the access/barrier is re-secured and verified.

Specific monitoring and radiological controls for access to "High Radiation Areas" are listed on the appropriate SWP.

As a minimum, each person entering a "High Radiation Area" shall have:

- DLR.
- Alarming direct reading dosimeter (electronic dosimeter).
- Stay time (if > 500 mrem per entry is expected).
- Approved SWP.
- Pre-job briefing on radiological conditions in the area.

RP Technicians should perform all briefings in High Radiation Areas using the CNS RP-800 Form for guidance.

RP personnel control access to HRAs for all personnel.

Personnel requesting such access will be equipped with one or more of the following:

A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

A radiation monitoring device, such as an electronic dosimeter, that has the capability to display accumulated dose and which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

A direct reading dosimeter and a qualified representative of the RP Department with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area, and who performs periodic radiation surveillance at the frequency specified for the applicable SWP; or

As specified in site Technical Specifications.

When a new, unanticipated HRA is discovered, then perform the following:

Ensure all personnel (if any) are immediately evacuated from the area and direct them to report to RP.

Guard or barricade the area to prohibit unauthorized access.

Maintain control of the area at all times.

Do not leave the area unguarded for any reason until proper procedural radiological controls have been established.

NOTE – Decisions regarding operation of plant equipment and systems are the sole responsibility of licensed Operations personnel.

Due to plant conditions, it may not be possible or advisable for Operations to implement requests for equipment status change.

If a request is made to Operations to secure equipment or a system lineup to address HRA conditions, then confirm action has occurred and verify HRA condition has been eliminated prior to leaving area unguarded.

Initiate a CR to document this occurrence.

LOCKED HIGH RADIATION AREA ACCESS CONTROL

All entrances or access points to Locked High Radiation Areas shall be locked with a distinct LHRA lock for the area or room.

These entrances shall remain locked except during periods of access by personnel under an approved SWP.

The following guidelines shall be used:
Lock each access to a LHRA; or
Establish an Access Control Guard to prevent unauthorized entry following the guidelines in Section; or
Control access to a LHRA through the use of SWPs, radiological postings, special door lock or padlock, and barricades to prevent unauthorized and inadvertent entries as required by Technical Specification 5.7.2 and Regulatory Guide 8.38.
If no enclosure exists for purposes of locking a LHRA located within a large area such as containment and an enclosure cannot be reasonably constructed, then:
Barricade and conspicuously post area.
Establish area radiation monitor with visual and audible signals set to alarm if radiation levels increase.
Alternate methods to control Locked High Radiation Areas (such as flashing lights) shall be approved by the Radiation Protection Manager or designee. Attachment 2, Approval For Locked High Radiation Area Deviations, may be used for documentation. Instruct personnel in the vicinity of these alternative controls as to their meaning and significance.

Specific monitoring and radiological controls for access to "Locked High Radiation Areas" shall be made by RP personnel and listed on the appropriate SWP. As a minimum, each person entering a "Locked High Radiation Area" shall have:

DLR.

Alarming direct reading dosimeter (electronic dosimeter).

Approved SWP.

RP Shift Technician or Radiation Protection Supervision (RPS) approval.

If an individual is working in a dose rate of ≥ 1000 mR/hr, then continuous RP coverage with the use of CNS RP-56, Radiological Stay Time Verification Sheet, is required.

Radiation Protection Manager or designee approval for entry into LHRAs with general area dose rates > 1.5 rem/hr in the actual work area.

Documented pre-job brief for entry, using the CNS RP-800, given by RP personnel. This brief shall cover radiological conditions in the immediate work areas using the most recent survey data available and the scope of the work to be performed.

RPS performs the pre-job brief for entry into the LHRAs with general dose rates > 1.5 rem/hr in the actual work area.

All briefings shall be performed by RP Supervision, ANSI 18.1, or ANSI 3.1 RP Technicians.

While LHRAs are open, the access to the LHRA shall be controlled per Technical Specifications.

Special Work Permit Area (SWP) - An area where an SWP has been issued to control access to, and work within, which involves any one or combination of the conditions: a Very High Radiation Area, **Locked High Radiation Area**, a High Radiation Area, High Contamination Area, Discrete Radioactive Particle Area, or an Airborne Radiation Area.

Distracters:

A. This answer is incorrect because continuous RP coverage is required. This answer is plausible because an SWP permit is required for entry and under some conditions Operations personnel can take action to protect the health and safety of the public without continuous RP coverage. The isolation of a small leak is not an instance where health and safety of the public is an issue. The candidate who believes continuous RP coverage is NOT required would select this answer.

C. This answer is incorrect because an SWP is required for entry. This answer is plausible because continuous RP coverage is required and under some conditions Operations personnel can take actions to protect the health and safety of the public without signing on a permit. However, the action to isolate a small leak is not an instance where the health and safety of the public is an issue. The candidate who understands continuous RP coverage is required but doesn't realize an SWP is required would select this answer.

D. This answer is incorrect because an SWP and Continuous RP coverage are required to enter a LHRA. This answer is plausible because there are instances where Operations personnel can take actions to protect the health and safety of the public without signing on a permit or signing on the SWP. The candidate who believes the action can be taken and doesn't realize the requirements of a high radiation entry would select this answer.

Technical Reference(s): Procedure 9.EN-RP-101, Access Control For Radiologically Controlled Areas, Rev 14
Proposed references to be provided to applicants during examination: none

Learning Objective: INT032-01-100 OPS CNS Administrative Procedures Radiation Protection

H. 9-EN-RP-101, Access Control for Radiologically Controlled Areas
1. Precautions and limitations
2. RCA access and egress

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (12)

Comments:

LOD 2

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.45
	Importance Rating	4.1

Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Question: 74

The Plant is at power when a transient occurs.

- Multiple Black and Yellow outlined annunciators are in

alarm. Which colored alarms take precedence and why?

A. BLACK; an EOP entry is required.

B. YELLOW; an EOP entry is required.

C. BLACK; a Plant Shutdown condition may be required.

D. YELLOW; a Plant Shutdown condition may be required.

Answer:
D. YELLOW; a Plant Shutdown condition may be required.

Explanation:

The window box assembly is a matrix of divided lamp windows with engraved legend plates and multi-colored window bezels. Each window has been given a "PRIORITY" signifying the importance of the alarm:

Priority I - RED; alarms that alert of EOP entry conditions or conditions requiring or causing an automatic or manual plant shutdown, or significant system setpoints.

Priority II - YELLOW; alarm conditions which may require or rapidly cause a plant shutdown or radiation release.

Priority III - BLACK; alarms that indicate off normal plant conditions that affect plant or component operability but should not lead to plant shutdown or radiation release.

Distracters:

A. This option is incorrect because black outlined alarms do not represent EOP entry conditions and do not have a higher priority than yellow outlines alarms. The candidate may choose this if he/she does not understand the color of EOP entry tiles. This option is plausible because black outlines alarms do indicate off normal conditions.

B. This option is incorrect because yellow outlined alarms do not indicate an EOP entry condition is present. The candidate may choose this if he/she does not understand the meaning of color-coded alarms. This option is plausible because yellow outlined alarms do indicate off normal conditions and do have a higher priority than black outline alarms.

C. This option is incorrect because the black outlined alarms do not indicate a plant shutdown condition may be present. The candidate may choose this if he/she does not understand the meaning of color-coded alarms. This option is plausible because black outlined alarms do indicate off normal conditions.

Technical Reference(s): SOP 2.2.64, Annunciator System, Rev. 17
AP 2.3.1, General Alarm Procedure, Rev. 62

Proposed references to be provided to applicants during examination: none

Learning Objective: COR002-35-02, Plant Annunciator System

LO-02 State the purpose of the following components related to the Plant Annunciator System.:

k. Alarm Window Boxes

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 3

Examination Outline Cross-Reference:

Level	RO
Tier #	3
Group #	
K/A #	2.4.49
Importance Rating	4.6

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
(CFR: 41.10 / 43.2 / 45.6)

Question: 75

There are operational circumstances when operators must perform immediate operator actions without reference to procedures.

Which statement represents one of those circumstances?

A. When an EOP directs performing the action.

B. When a scram is directed by Technical Specifications.

C. When an alarm procedure directs performing the action.

D. When an abnormal procedure directs performing the action.

Answer:

D. When an abnormal procedure directs performing the action.

Explanation:
Abnormal and Emergency (Non-EOP) procedures contain Immediate Operator Actions which the control room operator has committed to memory. Should a condition become true concerning the Immediate Operator Actions, Procedure 2.0.1.2 directs performing the action without use of procedures.

Distractors:

A: This answer is incorrect because EOPs do not contain immediate operator actions. Actions are taken per the EOPs without the control room operator having the procedure in hand, but the actions are directed by the CRS. This answer is plausible because actions are directed to be taken without use of the procedure but they are not immediately performed from memory. The candidate who recalls immediately performing actions directed from EOPs would select this answer.

B: This answer is incorrect because no actions are taken immediately from Technical Specifications. This answer is plausible because Technical Specifications have completion times to be taken immediately, but per TS immediate means to pursue without delay and in a controlled manner.

C: This answer is incorrect because alarm procedures do not contain immediate operator actions. This answer is plausible because actions can be performed immediately after entering the procedure. The candidate who recalls scram actions contained in alarm procedures may believe the action can be taken prior to entering the alarm procedure would select this answer.

Technical Reference(s): Procedure 2.0.1.2, Operations Procedure Policy, Rev. 44

Proposed references to be provided to applicants during examination: none

Learning Objective: INT032-01-03 (OPS CNS Administrative Procedure Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

G. Procedure 2.0.1.2, Operations Procedure Policy

1. Discuss the following as described in Procedure 2.0.1.2, Operations Procedure Policy.

f. Attachment 1, Immediate Operator Actions

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (10)

Comments:

LOD 2

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit:
Region: I ☐ II ☐ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results	
RO/SRO-Only/Total Examination Values	_____ / _____ / _____ Points
Applicant's Scores	_____ / _____ / _____ Points
Applicant's Grade	_____ / _____ / _____ Percent

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295006G2.1.6
Importance Rating	4.8

295006 SCRAM

2.1.6: Ability to manage control room crew during plant transients.

Question: 76

Which of the following completes the statement below identifying the CRS direction to the Reactor Operators following a non-complicated Automatic Scram from rated power and the procedure which provides guidance for this direction?

Assign the RO Procedure 2.1.5 (Reactor Scram) Attachments ____(1)____ IAW Procedure ____(2)____.

Attachment 1 Mitigating Task Scram Actions
Attachment 2 Reactor Power Control
Attachment 3 Reactor Water Level Control
Attachment 4 Reactor Pressure Control
Attachment 5 Balance of Plant Actions

A. (1) 1 & 2 ONLY AND assign the BOP Attachments 3, 4, and 5.
(2) 2.0.3 (Conduct of Operations)

B. (1) 1 & 2 ONLY AND assign the BOP Attachments 3, 4, and 5.
(2) 2.0.1.3 (Time Critical Operator Action Control and Maintenance)

C. (1) 1, 2 & 3 ONLY AND assign the BOP Attachments 4 and 5.
(2) 2.0.3 (Conduct of Operations)

D. (1) 1, 2 & 3 ONLY AND assign the BOP Attachments 4 and 5.
(2) 2.0.1.3 (Time Critical Operator Action Control and Maintenance)

Answer:

C. (1) 1, 2 & 3 ONLY AND assign the BOP Attachments 4 and 5.
(2) 2.0.3 (Conduct of Operations).

Explanation:
Requires knowledge and coordination of procedure 2.1.5 (Reactor Scram) Attachments and selecting the procedure which provides for division of operator responsibilities during a transient. IAW procedure 2.0.3, The CRO-RO is normally responsible for reactivity control, safe reactor shutdown, and mitigating task scram actions and post-scram reactor level control

(Attachments 1, 2, and 3 of Procedure 2.1.5). The CRO-BOP is normally responsible for post-scram Pressure Control and BOP System operation (Attachments 4 and 5 of Procedure 2.1.5). Assign the RO Procedure 2.1.5 (Reactor Scram) Attachments 1, 2, & 3 AND the BOP Attachments 4& 5 IAW Procedure 2.0.3 (Conduct of Operations).

Distracters:

A. This answer is incorrect due to RPV Level control being assigned to the RO. This answer is plausible if ATWS conditions were present, assigning level control to the BOP would be correct. The candidate who confuses reactor conditions following a scram and correctly identifies the procedure providing guidance would select this answer.

B. This answer is incorrect due to RPV Level control being assigned to the RO and the procedure not providing division of operator responsibilities. This answer is plausible if ATWS conditions were present, assigning level control to the BOP would be correct and if scram mitigating actions are confused with time critical operator action. The candidate who confuses reactor conditions following a scram and confuses time critical operator actions with division of operator responsibilities would select this answer.

D. This answer is incorrect due to the procedure not providing division of operator responsibilities. This answer is plausible if scram mitigating actions are confused with time critical operator action. The candidate who correctly identifies Attachments 1, 2, & 3 and confuses time critical operator actions with division of operator responsibilities would select this answer.

Technical Reference(s):
Procedure 2.0.3 (Conduct of Operations), Rev. 87
2.0.1.3 (Time Critical Operator Action Control and Maintenance), Rev. 02

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT032010400D0400 Describe the general sequence of events performed in the Reactor Scram section of procedure 2.1.5, Reactor Scram.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires coordination of procedure attachments and selection of procedure which provides this guidance.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295018AA2.03
Importance Rating	3.5

295018 Partial or Complete Loss of Component Cooling Water

AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

AA2.03: Cause for partial or complete loss

Question: 77

The plant is operating at 100% power when the following conditions occur:

- SW Pressure on both Divisions has risen but is still in the green band.
- REC system pressure is steady and is in the green band.
- REC Surge Tank Level High alarms.
- RR MG Set Oil Temperatures are rising.
- Drywell temperature and pressure are rising.
- RWCU F/D Inlet Temp High alarms.

Which one of the following identifies the cause of these conditions and the required action to correct the problem?

The REC Heat Exchanger…

A. REC Outlet valve has closed, shift REC heat exchangers IAW 5.2SW.

B. REC Outlet valve has closed, shift REC heat exchangers IAW 5.2REC.

D. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2SW.

C. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2REC.

Answer:
C. SW Outlet valve has closed, shift REC heat exchangers IAW 5.2REC.

Explanation:

A loss of SW flow to the REC Heat Exchanger recovery is covered in both 5.2REC and 5.2SW (assumes loss of SW pumps or piping). If the pressure of the service water system lowers to < 38 psig the system will isolate non-critical loads. The subsequent steps will have the Operators place the other loops REC Heat Exchanger in service. For the given condition, SW Pressure rising indicates that there was some restriction in the flow path. With REC temperatures rising and systems cooled by REC alarming, the flow restriction is affecting the REC Heat Exchanger. 5.2SW & 5.2REC direct shifting heat exchangers for different reasons. 5.2SW shift is to use a good SW loop vs. 5.2REC due to REC cooling issues.

Distracters:

A. This answer is incorrect because the REC outlet valve closing will not cause SW pressure to rise and 5.2SW not being the procedure utilized to shift heat exchangers under the provided conditions. This answer is plausible because closure of this valve would provide the other indications and if the stem were changed to reflect REC pressure rising would be correct and 5.2SW provides guidance to shift (use a good SW loop vs. 5.2REC due to REC cooling issues). The candidate who confuses indications provided and which procedure provides the correct guidance would select this answer.

B. This answer is incorrect because the REC outlet valve closing will not cause SW pressure to rise. This answer is plausible because closure of this valve would provide the other indications and if the stem were changed to reflect REC pressure rising would be correct along. The candidate who confuses indications provided and correctly identifies the procedure providing the correct guidance would select this answer.

D. This answer is incorrect because 5.2SW is not the procedure utilized to shift heat exchangers under the provided conditions. This answer is plausible because 5.2SW provides guidance to shift (use a good SW loop vs. 5.2REC due to REC cooling issues). The candidate who correctly identifies the cause and confuses which procedure provides the correct guidance would select this answer.

Technical Reference(s):
Emergency Procedure 5.2REC, Loss of REC, Rev. 16
Emergency Procedure 5.2SW, Service Water Casualties, Rev. 24.

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0320126L0L0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Question Source:

Bank #
Modified Bank # X (See attached)
New

Question History Last NRC Exam: 2011 Question 78

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 5

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires assessment of plant conditions and selection of procedure to mitigate the conditions.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 3 78		00	02/11/11		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
ABN / Emergency Procedures	Loss of SW flow to an REC heat exchanger, which procedure and why?

Related Lessons
INT032-01-26 Cooling Water Abnormal

Related Objectives
L. Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.2REC	Loss of Reactor Equipment Cooling
5.2SW	Service Water Casualties

Related Skills (K/A)
295018.AA203 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / **43.5** / 45.13) Cause for partial or complete loss (3.2 / 3.5)

QUESTION: S 3 78

The plant is operating at 100% power when the following conditions occur:

- SW Pressure on both Divisions have risen but are still in the green band
- REC system pressure is steady and is in the green band
- REC Surge Tank Level High alarms
- RR MG Set Oil Temperatures are rising
- Drywell temperature and pressure are rising
- RWCU F/D Inlet Temp High alarms

What is the cause and which procedure should be entered to correct the problem?

a. REC Heat Exchanger SW Outlet valve failed closed; enter 5.2REC to correct the problem.

b. Service water non-critical loop isolated; enter 5.2SW to correct the problem.

c. REC system has developed a leak; enter 5.2REC to correct the problem.

d. Service water pump tripped; enter 5.2SW to correct the problem.

ANSWER: S 3 78

a. REC Heat Exchanger SW Outlet valve failed closed; enter 5.2REC to correct the problem.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295019G2.4.4
Importance Rating	4.7

295019 Partial or Complete Loss of Instrument Air

2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question: 78

An air leak occurs with the plant operating at rated power:

(1) What indication requires entry into 5.2Air (Loss of Instrument Air)?
AND
(2) When is 5.2 AIR Attachment 2 (IA Pressure Loss) required to be implemented?

A. (1) If Instrument Air pressure ONLY remains below the green band.
(2) If Instrument Air pressure stabilizes at 84 psig.

B. (1) If Instrument Air pressure ONLY remains below the green band.
(2) If Instrument Air pressure stabilizes at 75 psig.

C. (1) If Service Air OR Instrument Air pressure remains below the green band.
(2) If Instrument Air pressure stabilizes at 84 psig.

D. (1) If Service Air OR Instrument Air pressure remains below the green band.
(2) If Instrument Air pressure stabilizes at 75 psig.

Answer:

D. (1) If Service Air OR Instrument Air pressure remains below the green band.
(2) If Instrument Air pressure stabilizes at 75 psig.

Explanation: Requires recognition of Emergency Procedure entry due to abnormal system parameter indications and evaluating plant conditions to determine when to implement attachments. 5.2AIR entry is required if SA or IA pressure is below green band and does not recover back into green band. Attachment 2 provides instructions that are performed when system pressure is considered to be too low to support continued operation ($\leq$ 77 psig)

Distracters:

A. This answer is incorrect due to IA low pressure not being the only entry condition and pressure stabilizing at 84 psig does not require Attachment 2 performance. IA low pressure ONLY is plausible due to procedure being titled IA pressure loss and 5.2AIR provides specific supplemental guidance for IA pressure < 85 psig. Candidates not knowing procedure entry conditions and air pressures which do not support continued operation would choose this answer.

B. This answer is incorrect because IA low pressure is not the only entry condition. IA low pressure ONLY is plausible because the procedure is titled IA pressure loss. Candidates not knowing procedure entry conditions, but knowing air pressures which do not support continued operation would choose this answer.

C. This answer is incorrect because pressure stabilizing at 84 psig does not require Attachment 2 performance. This pressure is plausible because 5.2AIR provides specific supplemental guidance for IA pressure < 85 psig. Candidates knowing entry conditions and not knowing the air pressures which do not support continued operation would choose this answer.

Technical Reference(s):
Procedure 5.2Air (Loss of Instrument Air), Rev. 19.

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0011702001070A Given a specific Plant Air system malfunction, determine the effect on any of the following: Plant Operation

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 2
55.45 6

Difficulty:

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
This meets SRO ONLY due to requiring knowledge of when to implement attachments associated with emergency procedures.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295023A2.05
Importance Rating	4.6

295023 Refueling Accidents

AA2 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS:

A2.05 Entry conditions of emergency plan

Question: 79

The plant is in MODE 5 with refueling in progress when an irradiated fuel bundle is dropped over the core.

The Refueling Floor ARM (RA-1) is 5.5 x 10^4 mR/hr and rising.
The ERP Kaman is indicating 1.80E +06 µCi/sec and rising.

(1) What is the current MINIMUM required Emergency Classification?
(2) What ERP Kaman reading requires escalation to the next higher classification?

A. (1) Unusual Event
(2) 3.70E+06 µCi/sec

B. (1) Unusual Event
(2) 3.70E+07 µCi/sec

C (1) Alert
(2) 3.70E+06 µCi/sec

D. (1) Alert
(2) 3.70E+07 µCi/sec

Answer:
D. (1) Alert
(2) 3.70E+07 µCi/sec

Explanation:
An Alert declaration is required if irradiated fuel is damaged and refuel floor radiation levels exceed 50 R/hr. If refuel floor ARM rising, combined with reactor cavity or spent fuel pool level lowering, the threshold for an Unusual Event declaration is met. The ERP Kaman indication is also a NOUE. The Site Area Emergency for release rate from the ERP is 3.50E+07 µCi/sec. With the information given, an Alert is required to be declared. Note 2 for the EAL Category A, states not to wait for the 15 minute time to elapse before declaring if the start time is unknown. No times were given in the question so this requirement is met.

Distracters:

A. This answer is incorrect because an Alert is required. A UE would be correct if radiation level was provided with a lowering SFP level. All other SAE values in Table A-1 are established at 3.50E+06 µCi/sec. Since 3.70E+06 is greater than this value, it is possible to consider the threshold being exceeded. The candidate who chooses an incorrect release point would select this answer. This answer is plausible because the UE threshold is met.

B. This answer is incorrect because an Alert is required. A UE would be correct if radiation level was provided with a lowering SFP level. 3.70E+07 is correct for this release point to require a higher EAL classification. The candidate who misses the damaged fuel EAL would select this answer. This answer is plausible because the UE threshold is met.

C. This answer is incorrect because the listed radiation release is too low for escalation to the next higher EAL. All other SAE values in Table A-1 are established at 3.70E+06 µCi/sec. Since 3.7E+06 is greater than this value, it is possible to consider the threshold being exceeded. If the candidate misreads the table, then this answer would be chosen. This answer is plausible because the release rates are contained in Table A-1.

Technical Reference(s):
Procedure EPIP 5.7.1 Attachment 4, (Emergency Action Level Matrix), Rev. 11
Procedure EPIP 5.7.1 (Emergency Classification), Rev. 50.

Proposed references to be provided to applicants during examination: EAL Matrix
Category A

Learning Objective:

GEN0030401C0C050E Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency classification.

Question Source:

Bank # 19335
Modified Bank #
New

Question History: 2011 NRC exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
The SRO is responsible for EAL classification declarations.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295028G2.4.11
Importance Rating	4.2

295028 High Drywell Temperature

2.4.11 Knowledge of abnormal condition procedures.

Question: 80

While operating at 80% power, the following indications are observed:

- Drywell Temperature is 135°F and rising slowly (0.25°F/min).
- Drywell Pressure is 0.55 psig and rising slowly (0.05 psig/min).

(1) Which action is required to be directed IAW 2.4PC (Primary Containment Control)?
AND
(2) Is 2.4PC, Attachment 1 (Primary Containment Relative Humidity) required to be utilized under these conditions?

A. (1) Ensure all available drywell FCU control switches in RUN.
(2) Yes

B. (1) Ensure all available drywell FCU control switches in RUN.
(2) No

C. (1) Ensure all available drywell FCU control switches in OVERRIDE.
(2) Yes

D. (1) Ensure all available drywell FCU control switches in OVERRIDE.
(2) No

Answer:

A. (1) Ensure all available drywell FCU control switches in RUN.
(2) Yes

Explanation: Requires knowledge of abnormal procedure 2.4PC supplemental actions due to high DW temperature. 2.4PC requires verification of drywell FCU control switches in RUN vs. EOP 3A which directs placing switches in OVERRIDE. If stem were changed to provide DW temperature > 150°F – OVERRIDE would be correct. Attachment 1 is required to be evaluated under these conditions since the cause of rising DW temperature and pressure is unknown. If cause was due to DW FCU trips (stem change) then reason is known and determining humidity is not required.

Distracters:

B. This answer is incorrect due to Attachment 1 being required to support determining the cause of rising DW parameters. This answer is plausible if the stem stated DW FCU(s) tripped which would make this answer correct.

C. This answer is incorrect due to placing FCU switches in OVERRIDE. This answer is plausible if EOP actions vs. AOP actions are confused. The candidate who confuses this action but correctly identifies Attachment 1 being required would choose this answer.

D. This answer is incorrect due to placing FCU switches in OVERRIDE and Attachment 1 being required to support determining the cause of rising DW parameters. This answer is plausible if EOP actions vs. AOP actions are confused and if the stem stated DW FCU(s) tripped which would make this answer correct.

Technical Reference(s):
Procedure 2.4PC (Primary Containment Control), Rev. 17.

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT0320128K0K0100 Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5
55.45 13

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
This meets SRO ONLY due to requiring knowledge supplemental AOP actions and when to implement attachments associated with abnormal procedures.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295005AA2.08
Importance Rating	3.3

295005 Main Turbine Generator Trip

EA2 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:

AA2.08 Electrical distribution status

Question: 81

The plant is operating at rated power with the Startup Transformer out of service for maintenance.

Which one of the following identifies:

(1) The impact to the electric plant if the Main Turbine trips?
AND
(2) Is the CRS required to direct DCC to perform the CNS- Black Plant Procedure?

A. (1) ALL 4160 VAC Buses de-energize until the Diesel Generators energize the Critical Buses.
(2) Yes

B. (1) ALL 4160 VAC Buses de-energize until the Diesel Generators energize the Critical Buses.
(2) No

C. (1) ALL 4160 VAC Buses de-energize until the Emergency Transformer energizes the Critical Buses.
(2) Yes

D. (1) ALL 4160 VAC Buses de-energize until the Emergency Transformer energizes the Critical Buses.
(2) No

Answer:

C. (1) ALL 4160 VAC Buses de-energize until the Emergency Transformer energizes the Critical Buses.
(2) Yes

Explanation: Requires knowledge of electrical distribution status following a Main Turbine Generator trip at power with the Startup Transformer out of service AND assessing plant conditions to determine if supplemental actions contained in Attachment 3 (Electrical Systems Guideline) of 5.3EMPWR (Emergency Power During Modes 1, 2, OR 3) are required.

Distracters:

A. This answer is incorrect due to the Critical Buses being energized from the DGs. This answer is plausible if the order in which emergency power supplies energize the Critical Buses is confused or if the stem were changed to reflect the emergency transformer being unavailable. The candidate who confuses the order in which emergency power supplies energize the Critical Buses and correctly knows to direct DCC to enter the Black Plant procedure would select this answer.

B. This answer is incorrect due to the Critical Buses being energized from the DGs and the Black Plant procedure entry not being required. This answer is plausible if the order in which emergency power supplies energize the Critical Buses is confused or if the stem were changed to reflect the emergency transformer being unavailable AND if the Black Plant procedure entry misinterpreted to only be required during a Station Blackout (SBO). The candidate who confuses the order in which emergency power supplies energize the Critical Buses and does not know to direct DCC to enter the Black Plant procedure IAW 5.3EMPWR would select this answer.

D. This answer is incorrect due to the Black Plant procedure entry not being required. This answer is plausible if the Black Plant procedure entry misinterpreted to only be required during a Station Blackout (SBO). The candidate who correctly identifies the order in which emergency power supplies energize the Critical Buses and does not know to direct DCC to enter the Black Plant procedure IAW 5.3EMPWR would select this answer.

Technical Reference(s):
Procedure 5.3EMPWR (Emergency Power During Modes 1, 2, OR 3), Rev. 48.
Procedure 5.3SBO (Station Blackout) Rev. 33

Proposed references to be provided to applicants during examination: None

Learning Objective: INT0320126Q0Q0100, Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 5
55.45 13

LOD 3

SRO Only - 10 CFR 55.43(b)(5) - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires assessing plant conditions to determine if supplemental actions contained in Attachment 3 (Electrical Systems Guideline) of 5.3EMPWR (Emergency Power During Modes 1, 2, OR 3) are required.

Examination Outline Cross-Reference:

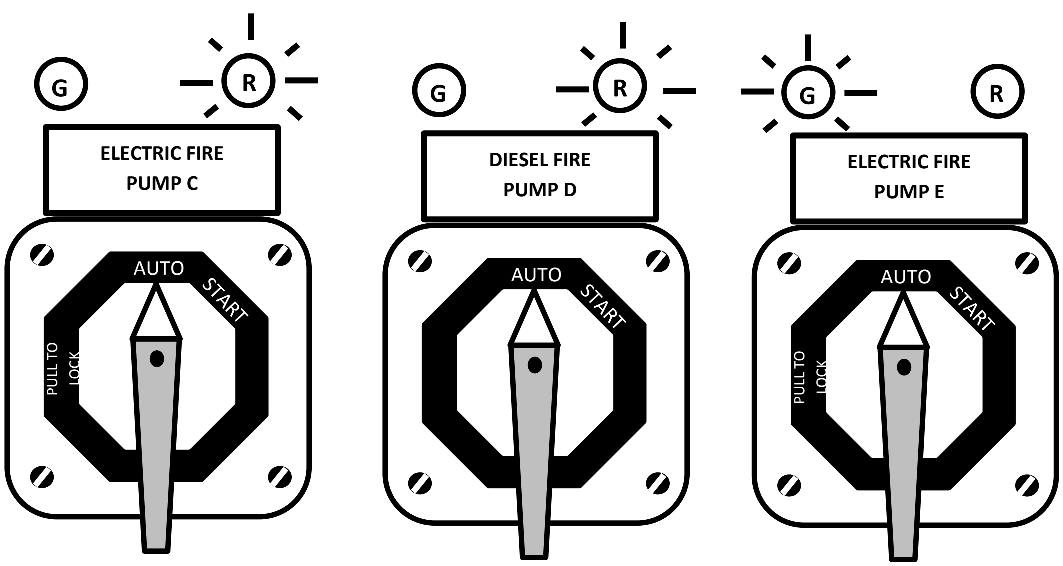
Level	SRO
Tier #	1
Group #	1
K/A #	600000G2.2.44
Importance Rating	4.4

600000 Plant Fire On Site

2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question: 82

While the Fire Brigade is combatting a Temporary Trailer fire (south of the Training Building) with fire hoses, the following indications are observed on the Main Control Room Fire Protection Panel:



The operator then places the control switch for Fire Pump E to START and the pump starts.

(1) What is the minimum pressure reached in the fire protection header prior to the operator taking the above action?

(2) What is the operability status of the Fire Suppression Water System?

A. (1) 68 psig
(2) Operable

B. (1) 68 psig
(2) Inoperable

C. (1) 141 psig

(2) Operable

D. (1) 141 psig
(2) Inoperable

Answer:
B. (1) 68 psig
(2) Inoperable

Explanation:
Requires candidate to determine Fire Protection header pressure which automatically starts ALL Fire Pumps and recognize failure of Fire Pump E to auto start (interpret control room indications). Fire Pump C starts at the lowest pressure of 68 psig. Per Surveillance Procedure 6.FP.101, Fire Pump 31 Day Operability Test, Attachment 2 Step 2.4, Electric Fire Pump C is tested to TRM surveillance requirements to justify as a backup to the Fire Suppression Water System and the steps are identified as Non-Technical Specification because Fire Pump C is not a TRM component. 141 psig is the start setpoint for Fire Pump D with a 10 second time delay. With this pump running this choice is plausible. Second part requires knowledge of TRM bases (requires Fire Pump E to support operability) and TSR 3.11.2.13 which requires each pump to sequentially start based upon lowering system pressure. Fire Pump E failed to auto start on low system pressure and is therefore Inoperable <u>even if manually started</u> (operator action impact) from the Control Room.

Distracters:

A. This answer is incorrect due to the Fire Suppression system being inoperable. This answer is plausible if the SR for this pump to auto start on low system pressure is not known or Fire Pumps D & E are required to support system operability. The candidate who knows fire pump auto start setpoints and does not know pump auto start on low system pressure is required for operability or which pumps are required for operability would choose this answer.

C. This answer is incorrect due to 141psig being above the auto start setpoint of Fire pump C and the Fire Suppression system being inoperable. This answer is plausible if the Fire pump auto start setpoint are confused (If the stem were changed to indicate Fire Pump C not running, 141 psig would be correct) and the SR for this pump to auto start on low system pressure is not known or Fire Pumps D & E are required to support system operability. The candidate who confuses fire pump auto start setpoints and does not know pump auto start on low system pressure is required for operability or which pumps are required for operability would choose this answer.

D. This answer is incorrect due to 141psig being above the auto start setpoint of Fire pump C. This answer is plausible if the Fire pump auto start setpoint are confused (If the stem were changed to indicate Fire Pump C not running, 141 psig would be correct). The candidate who confuses fire pump auto start setpoints and knows pump auto start on low system pressure is required for operability and which pumps are required for operability would choose this answer.

Technical Reference(s):
Technical Requirements Manual (TRM), Rev. 8/27/2014
Procedure 2.3_FP-4 (Fire Protection - Annunciator 4), Rev. 11.
Procedure 2.2.30 (Fire Protection System), Rev. 62
Surveillance Procedure 6.FP.102 (Annual Testing of Fire Pumps), Rev. 33
Surveillance Procedure 6.FP.101, (Fire Pump 31 Day Operability Test), Rev. 35
Procedure 0.23 (CNS Fire Protection Plan), Rev. 71

Proposed references to be provided to applicants during examination: TRM 3.11.2 spec only (no SRs)

Learning Objective:

COR00105020010200
Given condition(s) and/or parameters associated with the Fire Protection system, determine if related Technical Requirements Manual Limiting Condition for Operation are met.

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.4
55.43.2
55.45.12

LOD 3

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the TS and their bases.

Requires knowledge of Fire Suppression Water System surveillance requirements, bases and CNS Fire Protection Plan.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	2
K/A #	295007A2.01
Importance Rating	4.1

295007 High Reactor Pressure

AA2 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:

A2.01 Reactor pressure

Question: 83

The plant is operating in Mode 1, End-of-Cycle with reactor pressure being maintained at 1015 psig.

A pressure adjustment is made at 0815 on September 1st.
The RO observes stable pressure on the following indicators at the specified time:

Time 0817	RFC-PI-90A, RX PRESS is indicating 1015 psig.
Time 0818	RFC-PI-90C, RX PRESS is indicating 1020 psig.
Time 0819	RFC-PI-90B, RX PRESS is indicating 1025 psig.

Which one of the following identifies the LATEST time the reactor is required to be in MODE 3 IAW TS 3.4.10, Reactor Steam Dome Pressure if these conditions persist?

A. 2018

B. 2019

C. 2033

D. 2034

Answer:

D. 2034

Explanation:
Technical Specification LCO 3.4.10 requires the reactor steam dome pressure be ≤ 1020 psig when in Modes 1 and 2. Condition A requires the pressure to be restored within limits within 15 minutes. Condition B requires the unit to be in Mode 3 within 12 hours if Condition A cannot be met. 0819 + 15 minutes (Condition A) + 12 hours = 0834 +12 hours = 2034.

Distracters:

A. This answer is incorrect because 2018 does not incorporate the 15 min Condition A completion time from the time 1020 psig was exceeded. This answer is plausible if Condition A is not applied. The candidate who does not apply the 15 min completion time of Condition A would select this answer.

B. This answer is incorrect because 2019 does not incorporate the 15 min Condition A completion time from the time 1020 psig was exceeded. This answer is plausible if Condition A is not applied and LCO entry is confused with pressure at 1020 psig vs. >1020. The candidate who does not apply the 15 min completion time of Condition A and confuses the actual LCO entry pressure would select this answer.

C. This answer is incorrect because 2033 incorporates the 15 min Condition A completion time from the wrong time 1020 psig was exceeded. This answer is plausible if the LCO entry is confused with pressure at 1020 psig vs. >1020. The candidate who does apply the 15 min completion time of Condition A and confuses the actual LCO entry pressure would select this answer.

Technical Reference(s):
Technical Specifications LCO 3.4.10

Proposed references to be provided to applicants during examination: LCO 3.4.10

Learning Objective:
INT007-05-05, OPS Tech Specs 3.4 Reactor Coolant System (RCS)
3. Given a set of plant conditions that constitutes noncompliance with a section 3.4 LCO, determine the ACTIONS that are required.
10. From memory, in MODES 1 and 2, state the actions required in ≤ one hour if the reactor steam dome pressure LCO is not met (LCO 3.4.10).

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Application of Required Actions (Section 3) IAW with rules of application requirements (Section 1).

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	2
K/A #	295020G2.2.14
Importance Rating	4.3

295020 Inadvertent Containment Isolation

2.2.14 Knowledge of the process for controlling equipment configuration or status.

Question: 84

A loss of Division 1 RPSPP1A power occurs while operating at power. Troubleshooting the problem is about to commence.

(1) What is the response of the RWCU PCIV(s)?
(2) What procedure controls the configuration of the RWCU PCIV(s) during troubleshooting?

A. (1) ONLY RWCU-MO-15 (INBOARD ISOLATION VALVE) closes.
(2) 0.31(Equipment Status Control) is the controlling procedure.

B. (1) ONLY RWCU-MO-15 (INBOARD ISOLATION VALVE) closes.
(2) 0-CNS-WM-102 (Work Implementation and Closeout) is the controlling procedure.

C. (1) BOTH RWCU-MO-15 and RWCU-MO-18 (OUTBOARD ISOLATION VALVE) close.
(2) 0.31(Equipment Status Control) is the controlling procedure.

D. (1) BOTH RWCU-MO-15 and RWCU-MO-18 (OUTBOARD ISOLATION VALVE) close.
(2) 0-CNS-WM-102 (Work Implementation and Closeout) is the controlling procedure.

Answer:
C. (1) BOTH RWCU-MO-15 and RWCU-MO-18 (OUTBOARD ISOLATION VALVE) close.
(2) 0.31(Equipment Status Control) is the controlling procedure.

Explanation:
The loss of Div 1 RPS causes both RWCU PCIVs (Inboard and Outboard) to close due to loss of power to the Non-regenerative heat exchanger outlet temperature instrument. The loss of Div 2 RPS power only closes the OUTBD isolation valve for RWCU. Second part requires knowledge & selection of procedures which provide guidance for configuration control during troubleshooting activities following a plant transient. Procedure 0.31 provides direction of configuration control of Motor operated isolation valves.

Distracters:

A. This option is incorrect because both RWCU-MO-15 and 18 PCIVs close. The procedure controlling configuration is correct. This answer is plausible because generally an RPS bus de-energizing causes one division valve only to close. The candidate who does not recall the power supply to the Non-regenerative heat exchanger outlet temperature switch and that it solely causes both PCIVs to close may select this option.

B. This option is incorrect because both RWCU-MO-15 and 18 PCIVs close. The procedure controlling configuration is not correct. This answer is plausible because generally an RPS bus de-energizing causes one division valve only to close. The candidate who does not recall the power supply to the Non-regenerative heat exchanger outlet temperature switch and that it solely causes both PCIVs to close may select this option. The candidate who does not recall the procedure that controls configuration in this instance may select this answer.

D. This answer is incorrect because the controlling procedure is 0.31 not 0-CNS-WM-102 The correct valve response is listed. This answer is plausible because the correct valve response is listed. The candidate who does not correctly recall the proper configuration control procedure but does recall the correct valve response may select this answer.

Technical Reference(s):
Procedure 0.31(Equipment Status Control), Rev. 71.
Procedure 0-CNS-WM-102 (Work Implementation and Closeout), Rev. 01.
Procedure 7.0.1.7 (Troubleshooting Plant Equipment), Rev. 14.

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022102001080C Given a specific RPS malfunction, determine the effect on any of the following: c. PCIS
INT032010100H010D Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: System component checklist requirements

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.3
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires assessment of plant conditions and selection of procedure which provides guidance for configuration control during troubleshooting.

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	2
K/A #	295034EA2.01
Importance Rating	4.2

295034 Secondary Containment Ventilation High Radiation

EA2 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION:

EA2.01 Ventilation radiation levels

Question: 85

The plant is in Mode 1 and annunciator 9-4-1/E-4, RX BLDG VENT HI-HI RAD alarms due to a valid signal.

(1) What is the LOWEST radiation level which causes this alarm to actuate?
(2) What is the TS Bases for the allowable value of this instrument setpoint?

A. (1) 5 mR/hr
(2) Detect a steam leak in Secondary Containment.

B. (1) 5 mR/hr
(2) Detect gross fuel cladding failure.

C. (1) 10 mR/hr
(2) Detect a steam leak in Secondary Containment.

D. (1) 10 mR/hr
(2) Detect gross fuel cladding failure.

Answer:
D. (1) 10 mR/hr
(2) Detect gross fuel cladding failure.

Explanation:
Requires the candidate to first determine the annunciator setpoint (radiation level) for Rx Bldg Vent Hi-Hi Rad (10 vs.5 mR/hr which is the Hi Rad). The second part requires knowledge of TS 3.3.6.2 (Secondary Containment Isolation Instrumentation) Bases for the setpoint. The setpoint is based upon detecting gross fuel cladding failure.

Distracters:

A. This option is incorrect as the setpoint for the Reactor Building Vent HI Hi rad is 10 mr/hr NOT 5 mr/hr. The basis for the setpoint is to detect gross clad fuel failure and NOT a steam leak in Secondary Containment. This choice is plausible if the Hi Rad is confused with the Hi HI Rad setpoint (change stem to reflect Hi Rad annunciator) and steam leak in Secondary Containment would provide elevated radiation levels but is not the bases. A candidate may select this answer if he/she confuse the Hi setpoint of 5 mr/hr with the Hi Hi and believe the bases for the setpoint is a steam leak in Secondary Containment.

B. This option is incorrect as the setpoint for the Reactor Building Vent HI Hi rad is 10 mr/hr NOT 5 mr/hr. This choice is plausible if the Hi Rad is confused with the Hi HI Rad setpoint (change stem to reflect Hi Rad annunciator). A candidate may select this answer if he/she confuse the Hi setpoint of 5 mr/hr with the Hi Hi and knows the bases for the setpoint is gross fuel cladding failure.

C. This option is incorrect as the setpoint is to detect gross clad fuel failure and NOT a steam leak in Secondary Containment. This choice is plausible due to a steam leak in Secondary Containment would provide elevated radiation levels but is not the bases. A candidate may select this answer if he/she correctly identify the Hi Hi setpoint and believe the bases for the setpoint is a steam leak in Secondary Containment.

Technical Reference(s):
Procedure 2.3_9-4-1 (PANEL 9-4 - ANNUNCIATOR 9-4-1), Rev. 46.
Technical Specification LCO 3.3.6.2, Secondary Containment Isolation Instrumentation

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR00118020010200
Given condition(s) and/or parameters associated with the Radiation Monitoring System, determine if related Technical Specification, Technical Requirements Manual, and Off Site Dose Assessment Manual Limiting conditions for Operation are met.

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.3
55.45.13

LOD 4

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the TS and their bases.
The SRO is responsible for knowledge of TS Bases for the alarm setpoint.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	203000A2.04
Importance Rating	3.6

203000 Residual Heat Removal /Low Pressure Coolant Injection: Injection Mode

A2 Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.04 A.C. failures

Question: 86

The following conditions exist during a large break LOCA from rated power:

- No Off-Site is power available.
- DG1 is unavailable.
- RHR Pump D is unavailable.
- Core Spray Pump B is unavailable.
- Reactor Building is inaccessible.
- RPV Pressure is 35 psig and steady.

(1) Which RHR Loop is available for LPCI injection?
(2) What action is required if additional injection is required to assure adequate core cooling?

A. (1) A
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

B. (1) A
(2) Enter 5.3ALT-STRATEGY (Alternate Core Cooling Mitigating Strategies) and inject using Fire Protection to RHR.

C. (1) B
(2) Enter 5.3ALT-STRATEGY (Alternate Core Cooling Mitigating Strategies) and inject using Fire Protection to RHR.

D. (1) B
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

ANSWER:
A. (1) A
(2) Enter 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3) and use the Supplemental Diesel Generator (SDG) to re-energize 4160V 1F Bus.

Explanation:
RHR Loop A consists of RHR pumps A & C. RHR Loop B consists of RHR pumps B & D. Loss of Div 1 4160V Bus 1F de-energizes RHR pumps A & B and AC valves in Loop A. The only pump with power is RHR Pump C (Loop A) with RHR-MO-27A (Outboard Injection Valve) de-energized in the OPEN position which provides a LPCI flowpath to the RPV with RHR-MO25A (Inboard Injection Valve) being DC powered. No RHR pumps are available in RHR Loop B (RHR pump D unavailable and B with no power). The connection of Fire Protection to SW Emergency Core Flooding ties into RHR Loop A piping. This connection is only directed by EP 5.3ALT-STRATEGY but requires excessive time to align the systems. Loss of all off-site power with only DG2 supplying Div 2 4160V Bus 1G with a LOCA requires entry into EOP-1A (RPV Control) and 5.3EMPWR. 5.3EMPWR provides guidance RPV and Containment, Balance of Plant, and Electrical Systems guidance via Attachments 1, 2, & 3. 5.3EMPWR provides direction to energize Bus 1F with the SDG. Requires SRO to determine the most effective and timely means of gaining additional RPV injection under conditions challenging Adequate Core Cooling. Utilizing the SDG will provide 2 additional RHR pumps for injection. EOP-1A provides guidance to utilize procedure 5.3ALT-STRATEGY to align required/desired systems, but due to extreme system/plant conditions, normal operation is not possible. RCIC is the preferred injection system, but with RPV at 35 psig RCIC will not provide sufficient makeup (injecting fire protection is the next preferred).

Distracters:

B. This answer is incorrect because aligning Fire Protection to RHR will not add water due to RHR system pressure being much higher than Fire Protection header pressure while RHR Pump C is operating. This choice is plausible because the alignment of FP can be performed. The candidate who correctly identifies the available RHR loop and confuses the differences of system driving head for injection would choose this answer.

C. This answer is incorrect because B Loop RHR is unavailable and the reactor building is inaccessible to manually open crosstie to Loop B (MO-20). This choice is plausible due to the diversity of power supplies to the RHR pumps & valves and 5.3ALT-STRATEGY provides guidance for Fire Protection connection to RHR but this will not work due to RHR system pressure and Fire Protection pressure differences. The candidate who incorrectly identifies the available RHR loop and confuses the Service Water availability for injection would choose this answer.

D. This answer is incorrect because B Loop RHR is unavailable. This choice is plausible due to the diversity of power supplies to the RHR pumps & valves. The candidate who incorrectly identifies the available RHR loop and correctly identifies the need to energize Bus 1F for the SDG would choose this answer.

Technical Reference(s):
Procedure 5.8 Attachment 1(RPV Control EOP-1A), Rev. 17.
Procedure 5.3ALT-STRATEGY Alternate Core Cooling Mitigating Strategies, Rev. 42.
Procedure 5.3EMPWR (Emergency Power During MODES 1, 2, OR 3), Rev. 43.

Proposed references to be provided to applicants during examination: None

Learning Objective:

COR0022302001080A Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Question Source:

Bank #
Modified Bank # 4029 (Attached)
New

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5
55.45.6

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires assessment of conditions and selection of procedure with which to proceed.

QUESTION: 172 4029 (1 point(s))

Given the following plant conditions :

- The plant was at 100% power
- The breaker for RHR Pump 1D was tagged out for maintenance
- A loss of 4160 VAC Switchgear Critical Bus 1F occurs
- Access to the Reactor Building is prohibited.

How is operation of RHR <u>from the Control Room</u> affected?

Only one RHR pump is available in RHR ...

a. Loop A.
Containment Cooling and LPCI injection can only be established on RHR Loop A.

b. Loop A.
Containment Cooling cannot be established in either loop.
LPCI injection is available, if needed.

c. Loop B.
Containment Cooling and LPCI injection can only be established on RHR Loop B.

d. Loop B.
Containment Cooling cannot be established in either loop.
LPCI injection is available, if needed.

ANSWER: 172 4029

b. Loop A.
Containment Cooling cannot be established in either loop.
LPCI injection is available, if needed.

Explanation of answer:

C RHR pump has power. Most of the AC valves in A RHR loop have no power. The SWB pumps in A RHR loop have no power. RHR-MO25A is DC powered. No RHR pumps in B RHR loop.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	212000G2.1.20
Importance Rating	4.6

212000 Reactor Protection System

2.1.20 Ability to interpret and execute procedure steps.

Question: 87

The plant is in Mode 5 with all OPERABLE Control Rods (Core Cell contains fuel) fully inserted.

Procedure 4.5 (Reactor Protection/Alternate Rod Insertion Systems), Section 7 (Bypass Reactor Mode Switch - Shutdown Position Scram and transfer MODE switch), directs the following Step:

7.3.2.2 Inform Shift Manager that Reactor Mode Switch - Shutdown Position Scram (LCO 3.3.1.1, Function 10) is inoperable due to being bypassed.

a. Enter applicable Conditions and Required Actions for following LCOs as required:

1. LCO 3.3.1.1.
2. LCO 3.10.3.
3. LCO 3.10.4.

What is the MINIMUM Required Action(s) for this step IAW TS LCO 3.3.1.1 (RPS Instrumentation) if an OPERABLE Control Rod is fully withdrawn with the DIV 1 Reactor Mode Switch Scram bypass jumper installed?

Place the bypassed channel in trip...

A. within 12 hours ONLY.

B. OR Insert the Control Rod within 1 hour.

C. OR Insert the Control Rod within 6 hours.

D. OR Insert the Control Rod within 12 hours.

Answer:
B. OR Insert the Control Rod within 1 hour.

Explanation:
Requires the step to be interpreted and executed to determine the TS Required actions. Requires knowledge of what is being bypassed by the jumpers (ONLY Mode Switch SHUTDOWN position is in RPS Channels A3 & B3). LCO is applicable in MODEs 1, 2 and 5 (With any control rod withdrawn from a core cell containing one or more fuel assemblies). Requires application of multiple TS ACTIONS to determine correct action with knowledge of TS 3.3.1.1 Bases (Loss of trip capability). Condition C requires restoration of RPS trip capability within 1 hour. If not completed within 1 hr, enter Condition H (as directed by TABLE 3.3.1.1-1) which requires initiating action to insert the Control Rod immediately. Options are to exit the MODE of applicability (all Control Rods inserted), place bypassed channel in trip or remove jumper to restore trip capability. With the bypassed channel in trip, trip capability is restored.

Distracters:

A. This answer is incorrect due to placing the bypassed channel in trip within 12 hours not being the only additional TS Action. This choice is plausible if loss of function in one channel and RPS trip (full scram) capability is not recognized and not understanding the impact of inserting the withdrawn control rod (no longer in the MODE of applicability). The candidate who does not recognize a loss of function & trip capability would choose this answer.

C. This answer is incorrect due to inserting the Control Rod within 6 hours not being the only additional TS Action. This choice is plausible if loss of function in one channel and RPS trip (full scram) capability is not recognized and not understanding the impact of inserting the withdrawn control rod (no longer in the MODE of applicability). The candidate who does not recognize a loss of RPS trip capability would choose this answer. Placing the bypassed channel in trip OR Inserting the Control Rod within 6 hours is plausible if both jumpers were installed with failure to recognize loss of trip (full scram) capability.

D. This answer is incorrect due to inserting the Control Rod within 12 hours not meeting the additional TS Action. This choice is plausible if loss of function in one channel and RPS trip (full scram) capability is not recognized and not understanding the impact of inserting the withdrawn control rod (no longer in the MODE of applicability). The candidate who does not recognize a loss of function & RPS trip capability but does understand exiting the Mode of applicability (insert Control Rod) would choose this answer.

Technical Reference(s):
Procedure 4.5 (Reactor Protection/Alternate Rod Insertion Systems), Rev. 31.
LCO 3.3.1.1 RPS Instrumentation
LCO 3.10.4 Single Control Rod Withdrawal--Cold Shutdown

Proposed references to be provided to applicants during examination: LCO 3.3.1.1 Table with Mode Switch info only

Learning Objective:

COR00221020010200
Given conditions and/or parameters associated with the RPS, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operations are met.
COR0022102001050D
Briefly describe the following concepts as they apply to RPS: Mode switch position

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.12

LOD 4

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the TS and their bases.
Requires application of TS action statements with knowledge to TS Bases. The SRO is required to determine "loss of function" for a given equipment condition.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	215003A2.05
Importance Rating	3.5

215003 Intermediate Range Monitor System

A2 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.05 Faulty or erratic operation of detectors/system

Question: 88

A reactor startup is in progress with the following conditions present:

- IRM G is Inoperable and bypassed.
- Reactor power is below the point of adding heat.

The following annunciator alarms and clears multiple times within 30 seconds and repeats for 3 minutes due to IRM A spiking.

IRM RPS CH A UPSCALE TRIP OR INOP	PANEL/WINDOW: 9-5-1/D-7

(1) What is the result of IRM A spiking?

(2) What is the required Technical Specification action?

A. (1) Rod block ONLY
(2) Place Channel or Associated Trip system in Trip within 12 hours.

B. (1) Rod block and 1/2 scram
(2) Place Channel or Associated Trip system in Trip within 12 hours.

C. (1) Rod block ONLY
(2) Place Channel in one trip system OR one trip system in trip within 6 hours.

D. (1) Rod block and 1/2 scram
(2) Place Channel in one trip system OR one trip system in trip within 6 hours.

Answer:
B. (1) Rod block and 1/2 scram
(2) Place Channel or Associated Trip system in Trip within 12 hours.

Explanation: Procedure 4.1.2 provides Abnormal IRM Readings on Attachment 1 to determine IRM operability due to Spiking. Since Rod Block & Scram setpoints (102.5 & 117.5/125 of scale) are reached (≥59/125 scale) IRM A is inoperable. The Trip System is made up of 4 Channels (IRMs A, C, E, and G are in Division 1 and B, D, F, and H are in Division 2). TS 3.3.1.1 Condition A is entered for IRMs A and G being inoperable. Table 3.3.1.1-1 requires 3 operable IRMs per trip system. With IRM A &G Inoperable, Table 3.3.1-1 Required Channels per Trip System requirements are not met. LCO 3.3.1.1, Condition A applies and must be met. If the IRMs were in different trip systems, then Condition B would apply.

Distracters:

A. This answer is incorrect because an RPS trip would also occur. The required action is correct. This answer is plausible because an IRM does cause a Rod Block and the required action is correct. The candidate who does not realize the RPS trip occurs would select this answer.

C. This answer is incorrect because and RPS trip would also occur and the listed required action is incorrect. If the candidate did not realize the IRMs were in the same trip system may select this answer based upon the required action. The 6 hours to place the channel in one trip system or one trip system in trip is the action taken if the IRMs were in different trip systems (e.g. Div 1 and Div 2). This answer is plausible because the listed required action is taken when its conditions are met.

D. This answer is incorrect because the listed required action is not correct. The system response is correct. If the candidate did not realize the IRMs were in the same trip system may select this answer. The 6 hours to place the channel in one trip system or one trip system in trip is the action taken if the IRMs were in different trip systems (e.g. Div 1 and Div 2). This answer is plausible because the system response is correct and the listed required action is an action one would take if the conditions were met.

Technical Reference(s):
Procedure 4.1.2 (Intermediate Range Monitoring System), Rev. 21.
Procedure 4.5 (Reactor Protection/Alternate Rod Insertion Systems), Rev. 31.
TS LCO 3.3.1.1, Reactor Protection System Instrumentation

Proposed references to be provided to applicants during examination: LCO 3.3.1.1

Learning Objective:

COR00212020010200
Given conditions and/or parameters associated with the IRM's, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operation are met.
COR0021202001090A
Given plant conditions, determine if the following IRM actions should occur: Rod Block.
COR0021202001090B
Given plant conditions, determine if the following IRM actions should occur: Reactor Scram.
INT007-05-05, OPS CNS Tech Specs 3.3, Instrumentation

3. Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41.50
55.43.2

LOD 3

SRO Only - Facility operating limitations in the technical specifications and their bases. (10 CFR 55.43(b)(2)).
The SRO is responsible for TS LCO required action determination.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	259002G2.2.40
Importance Rating	4.5

259002 Reactor Water Level Control System

2.2.40 Ability to apply Technical Specifications for a system

Question: 89

The Plant is operating at rated power on March 10th.

At 1300 the RO observes the following:

- RFC-LI-94A indicates 36 inches and is slowly rising.
- RFC-LI-94B indicates 32 inches and is slowly lowering.
- RFC-LI-94C indicates 37 inches and is slowly rising.
- Feedwater flow is slowly rising.

What is the MAXIMUM time allowed before reactor power must less than 25% RTP IAW TS 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation, if these conditions persist?

A. March 10 at 1500

B. March 10 at 1900

C. March 17 at 1300

D. March 17 at 1700

Answer:
D. March 17 at 1700

Explanation: Requires knowledge of TS instrument surveillance requirements and application of required actions associated with the RVLC System. The RVLC system utilizes NR level and FW flow instruments to properly control RPV water level. SR 3.3.2.2.1 requires a channel check which is documented in 6.LOG.601 requiring these Narrow Range level indicators to indicate within 2" of each other. Since the B NR is greater than 2" from A & C, this channel is considered inoperable (failed channel check). Condition A is entered at the time of failure and the required time to place the channel in a tripped condition is 7 days. Once the completion time is expired, reactor power must be lowered below 25% RTP IAW Condition C within 4 hours. 7 days + 4 hours = March 17 at 1700. If FW flow was changed to indicate lowering, A & C instruments would be considered inoperable.

Distracters:

A. This answer is incorrect due to only 1 NR instrument being inoperable. This answer is plausible due to 2 NR level indicators being higher than normal setpoint (35”) and reflects a 2 hour completion time. The candidate who incorrectly determines 2 level NR Level instruments are inoperable and confuses completion time to restore trip capability vs. reducing power to <25% would select this answer.

B. This answer is incorrect due to only 1 NR instrument being inoperable. This answer is plausible due to 2 NR level indicators being higher than normal setpoint (35”) and reflects a 2 + 4 hour (6 hrs) completion time. The candidate who incorrectly determines 2 level NR Level instruments are inoperable but correctly identifies the additional 4 hours to reducing power to <25% would select this answer.

C. This answer is incorrect due not applying the 4 additional hours allowed to reduce power below 25%. This answer is plausible due to 1 NR level indicator being lower than normal setpoint (35”) with higher FW flow and reflects the 7 day completion time (if stem were changed to ask when required to be in trip – would be correct). The candidate who correctly determines 1 level NR Level instrument is inoperable but confuses placing the channel in trip vs. reducing power to <25% would select this answer.

Technical Reference(s):
Technical Specifications LCO 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation.

Proposed references to be provided to applicants during examination: TS 3.3.2.2

Learning Objective:
INT007-05-04, OPS CNS Tech Specs 3.3, Instrumentation

3. Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.2
55.43.5
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases.
Requires assessing instrument / plant response and application of TS required actions.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	262001A2.06
Importance Rating	2.9

262001 A.C. Electrical Distribution

A2 Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Deenergizing a Plant Bus

Question: 90

The plant is in Mode 3 with DG2 Inoperable for the last 24 hours.

An event requires IMMEDIATELY de-energizing 4160V Bus 1A.

(1) Which one of the following identifies the impact of de-energizing Bus 1A immediately (Critical Bus is not manually transferred to the ESST)?
AND
(2) What is the required TS Required Actions once de-energized?

A. (1) A momentary loss of Bus 1F occurs.
(2) Restore Offsite Circuit or DG operability within 24 hours.

B. (1) A momentary loss of Bus 1F occurs.
(2) Restore Offsite Circuit within 7 days.

C. (1) A momentary loss of Bus 1G occurs.
(2) Restore Offsite Circuit or DG operability within 24 hours.

D. (1) A momentary loss of Bus 1G occurs.
(2) Restore Offsite Circuit within 7 days.

Answer:
A. (1) A momentary loss of Bus 1F occurs.
(2) Restore Offsite Circuit or DG operability within 24 hours.

Explanation:
Requires determination of Plant MODE to identify TS 3.8.1 applicability and applying TS required actions for multiple AC Source inoperability. Conditions provide plant in Mode 3. De-energizing Bus 1A without manually transferring Bus 1F results in 1 second time delay for auto transfer to the ESST (Impact = short duration de-energization). De-energizing Bus 1A results in loss of 1 offsite circuit. With DG2 already inoperable (TS 3.8.1 Condition B - 7 day LCO), 1 offsite circuit inoperable requires entry into Condition A (7 days) & Condition D (24 hours to restore DG or Offsite circuit).

Distracters:

B This answer is incorrect due to restoration of offsite circuit or DG being required within 24 hours. This choice is plausible if only the offsite circuit is considered inoperable. The candidate who correctly recognizes momentary loss of power to Bus 1F and does not recognize concurrent offsite circuit & DG inoperability would select this answer.

C. This answer is incorrect due to Bus 1G not being impacted by de-energizing Bus 1A. This choice is plausible if electric plant alignment is not known or confused. The candidate who confuses momentary loss of power to Bus 1F and correctly identifies concurrent offsite circuit & DG inoperability would select this answer.

D. This answer is incorrect due to Bus 1G not being impacted by de-energizing and Bus 1A restoration of offsite circuit or DG being required within 24 hours. This choice is plausible if electric plant alignment is not known or confused and only the offsite circuit is considered inoperable. The candidate who confuses momentary loss of power to Bus 1F and does not recognize concurrent offsite circuit & DG inoperability would select this answer.

Technical Reference(s):
Procedure 2.2.18 (4160V Auxiliary Power Distribution System), Revision 164.
TS LCO 3.8.1

Proposed references to be provided to applicants during examination: LCO 3.8.1

Learning Objective:
INT00705090010300 Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.5
55.45.6

LOD 3

SRO Only - 10 CFR 55.43(b)(2) - Facility operating limitations in the TS and their bases.
Requires application of TS Required Actions.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	2
K/A #	201006G2.1.23
Importance Rating	4.4

201006 Rod Worth Minimizer System

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question: 91

Which one of the following completes the statements below regarding how the “Select Error” function of the RWM is performed IAW 6.RWM.301 (Rod Worth Minimizer Functional Test For Startup) AND the significance of maintaining the RWM operable during startup IAW TS Bases?

The operator is required to verify the SELECT ERROR indicator turns red while selecting ____(1)____ from each RWM group (except Group 1).

The RWM enforces compliance with BPWS which ensures that the initial conditions of the analysis for a ____(2)____ is NOT violated.

A. (1) a single rod
(2) control rod drop accident

B. (1) a single rod
(2) single control rod withdrawal error

C. (1) ALL individual rods
(2) control rod drop accident

D. (1) ALL individual rods
(2) single control rod withdrawal error

Answer:

A. (1) a single rod
(2) control rod drop accident

Explanation: Requires knowledge of RWM Functional Test and TS 3.3.2.1 Bases. 6.RWM.301 (Rod Worth Minimizer Functional Test For Startup) requires the operator to verify the SELECT ERROR indicator turns red each time while selecting a single rod from each RWM group (except Group 1) and the RWM enforces BPWS which ensures that the initial conditions of the CRDA analysis are not violated.

Distracters:

B. This answer is incorrect due to a single control rod withdrawal error (RWE) not being the TS bases for the RWM. This choice is plausible due to a CRDA being easily confused with a RWE which is the bases for the Rod Block Monitor. The candidate who correctly identifies selecting only 1 rod in the remaining groups and confuses RBM vs. RWM bases would select this choice.

C. This answer is incorrect due to selecting all rods in the remaining groups not being required to support SELECT ERROR function verification. This choice is plausible due having many groups requiring single rod selection being easily confused with all the rods within each group. The candidate who confuses only 1 rod in the remaining groups vs. all rods and correctly identifies the RWM bases would select this choice.

D. This answer is incorrect due to selecting all rods in the remaining groups not being required to support SELECT ERROR function verification and a single control rod withdrawal error (RWE) not being the TS bases for the RWM. This choice is plausible due having many groups requiring single rod selection being easily confused with all the rods within each group and a CRDA being easily confused with a RWE which is the bases for the Rod Block Monitor. The candidate who confuses only 1 rod in the remaining groups vs. all rods and confuses RBM vs. RWM bases would select this choice.

Technical Reference(s):
Procedure RWM.301 (Rod Worth Minimizer Functional Test For Starup), Rev. 10.
TS Bases

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT007-05-04:
2. Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.2
55.45.6

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires knowledge of TS Surveillance Requirements and Bases. It is the SRO who determines how much of a system is required to be tested. In this case, the SRO must know that testing only the first control rod of each RWM group must be tested.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	2
K/A #	202002A2.06
Importance Rating	3.3

202002 Recirculation flow Control System

A2 Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Low reactor water level: Plant-Specific

Question: 92

The plant is operating in MODE 1 at 97% power and the following occurs:

- RFP A trips.
- RPV level lowers to 25 inches on NR Level instruments.

(1) How is the Recirculation Flow Control System affected?
AND
(2) What action is required?

A. (1) Both Recirculation Pumps run back to 22% speed.
(2) Direct the RO to scram the reactor IAW alarm 9-5-2/G-1 (Reactor Low Water Level).

B. (1) Both Recirculation Pumps run back to 22% speed.
(2) Direct the RO to ensure each Condensate Pump flow is within 5000 to 6000 gpm IAW 2.4 MC-RF (Condensate and Feedwater Abnormal).

C. (1) Both Recirculation Pumps run back towards 45% speed.
(2) Direct the RO to scram the reactor IAW alarm 9-5-2/G-1 (Reactor Low Water Level).

D. (1) Both Recirculation Pumps run back towards 45% speed.
(2) Direct the RO to ensure each Condensate Pump flow is within 5000 to 6000 gpm IAW 2.4 MC-RF (Condensate and Feedwater Abnormal).

Answer:
D. (1) Both Recirculation Pumps run back towards 45% speed.
(2) Direct the RO to ensure each Condensate Pump flow is within 5000 to 6000 gpm IAW 2.4 MC-RF (Condensate and Feedwater Abnormal).

Explanation:
If both reactor recirculation pumps are running and not locked out, RR pumps run back towards 45% speed if the following condition is met:

Total steam flow > 9 Mlbm/hr with at least 1 RFP tripped/flow < 1 Mlbm/hr and selected reactor water level < 27.5 inches. RR runback towards 45% stops when the condition causing the runback is no longer true and no other 45% runback conditions exist. With power at 97%, FW flow is around 9.27 Mlbm/hr. As the RR pumps start running back FW flow will lower below 9 Mlbm/hr and runback will stop. If the stem were changed to reflect Discharge valve closure or FW Flow <20% - 22% runback would be correct. If stem were changed to reflect level continuing to lower following runback – Reactor Low Level annunciator would be correct. On a trip of a RFP, entry into Abnormal Procedure 2.4MC-RF, Condensate And Feedwater Abnormal is required. Subsequent operator actions are to maintain main condensate flow of 5000 to 6000 gpm per running condensate pump. The CRS directs the RO to obtain this flow.

A. This answer is incorrect because both RR pumps run back towards 45% speed. Once the condition that caused the runback (lowering FW flow) is clear the RR pumps stop running back. There is no requirement to run back to 22% speed with initial reactor power at 97%. This answer is plausible because the RR pumps do have a 22% runback feature. The second part of the answer is incorrect because there is guidance to scram the reactor with level lowering.

B. This answer is incorrect because there is no reason to scram the reactor. If reactor level cannot be maintained above 12 inches, then the scram would be required. With level at 25 inches RPV level can be maintained above 12 inches. Both RR pumps will run back towards 45% speed. Once the condition that caused the runback (lowering FW flow) is clear the RR pumps stop running back. This answer is plausible because to RR response is correct. The candidate who recognizes the correct RR speed response and is not sure of the plant response at this power level may select this answer.

C. This answer is incorrect because both RR pumps run back towards 45% speed. Once the condition that caused the runback (lowering FW flow) is clear the RR pumps stop running back. There is no requirement for the pumps to run back to 22% speed with initial reactor power at 97%. This answer is plausible because the RR pumps do have a 22% runback feature. The candidate who is not sure of the RR response but reasons lowering reactor power is a prudent action to take may select this answer.

Technical Reference(s):
Procedure 2.2.68.1, Reactor Recirculation System Operations, Rev. 76.
ARP 2.3_A-1, Panel A – Annunciator A-2, Rev. 40.
ARP 2.3_9-5-2, Panel 9-5 – Annunciator 9-5-2, Rev. 43.
Procedure 2.4MC-RF, Condensate and Feedwater Abnormal, Rev. 14
Procedure 2.4RXLVL, RPV Water Level Control Trouble, Rev. 25

Proposed references to be provided to applicants during examination: None

Learning Objective:

4. Describe the interrelationships between the Reactor Recirculation system or the Recirculation Flow Control system and the following:
   j. Reactor water level/pressure
10. Describe the Reactor Recirculation system and/or Recirculation Flow Control system design features and/or interlocks that provide for the following:
   l. Recirculation pump runback
13. Given plant conditions, determine if any of the following should occur:
   c. Recirculation pump runback to the 45% speed limiter

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 41.5
45.6

LOD 4

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
The SRO assesses plant conditions and determines whether RPV level will or will not recover and make priority decisions about responding to a plant transient.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	2
K/A #	2710002.4.20
Importance Rating	4.3

271000 Offgas System

2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

Question: 93

The following conditions exist following an earthquake:

- The reactor is scrammed all control rods fully insert.
- No SRVs were able to be opened for pressure control.
- No means of RPV injection are available.
- RPV water level is -150 inches corrected fuel zone and steady.
- One MSL remains open.
- RPV pressure is 950 psig.
- Primary Containment pressure is 0.45 psig.
- Torus water level is 9.5 feet and lowering 0.2 feet/minute.
- The TSC is not operational

The CRS intent is to begin using the Steam Jet Air Ejectors (SJAEs).

(1) What is the CRS' response?
(2) What CAUTION is of concern when carrying out this response?

A. (1) Transition to EOP 2A and direct Emergency Depressurization.
(2) When placing SJAEs in service, caution of sending personnel through potentially high radiation areas must be addressed.

B. (1) Transition to EOP 2A and direct Emergency Depressurization.
(2) When placing SJAEs in service, forcing the shutter slide on the breaker could cause damage to the breaker.

C. (1) Transition to EOP 2A and direct Steam Cooling.
(2) When placing SJAEs in service, caution of sending personnel through potentially high radiation areas must be addressed.

D. (1) Transition to EOP 2A and direct Steam Cooling.
(2) When placing SJAEs in service, forcing the shutter slide on the breaker could cause damage to the breaker.

Answer:
A. (1) Transition to EOP 2A and direct Emergency Depressurization.
(2) When placing SJAEs in service, caution of sending personnel through potentially high radiation areas must be addressed.

Explanation:
The CRS must direct Emergency Depressurization due to EOP 3A direction on torus water level being below 9.6' of water. When the downcomers become uncovered, the drywell and torus pressure suppression function is lost because any possible future steam leakage into the drywell will not be directed below the torus water level and condensed. The result could be primary containment failure due to overpressure. Emergency depressurization is the primary action to take with the given conditions. To accomplish the ED, the steam jet air ejectors are going to be used. The CRS is knowledgeable of all the plant conditions and can prioritize which system(s) is/are to be used and the priority placed on the order to use the systems. EOPs allow using one or all of the listed systems and the CRS directs placing as many in service that is required to perform the task of lowering RPV pressure less than 50 psig below tours space pressure. The CRS must be knowledgeable about the CAUTIONS applicable to which system to use and prioritize based upon procedure guidance and plant conditions. At this point ED is complete and the RPV will not again pressurize.

Distracters:

B, This answer is incorrect because the CAUTION of concern is not correct. There are no breaker trip actions to take when placing the SJAEs in service. This caution is based upon placing the AOG third stage SJAEs in service which is contained in the same procedure. The guidance is contained in the section of EOP procedure 5.8.1 which is knowledge the SRO is required to know but is not general RO knowledge based upon the placement being well inside the procedure guidance. This answer is plausible because the action to transfer to EOP 2A and emergency depressurize is required. The second part is plausible because it is a caution that is in the procedure for placing a like system in service.

C. This answer is incorrect because the requirement to steam cool is not met. RPV level is steady and as long as level can be maintained above -183 inches with no means of injection available. The second part of the answer is correct because there is a potential of sending personnel through a high radiation field when performing the task. It is the SROs responsibility to determine if personnel safety risk merits performing the task. The SRO has determined it safe as inferred in the question stem. This answer is plausible because the correct caution is stated

D. This answer is incorrect because both parts of the answer are incorrect. The correct action is to emergency depressurize and the correct caution to regard is sending the personnel through the potential high radiation field. This answer is plausible because the actions given are correct if given different circumstances. The same procedure is used for using the AOG third stage SJAEs in service and this system can be used as an emergency depressurization system. The SRO must know detailed procedure guidance to realize there are no breaker trips with the slide shutter when using the SJAEs.

Technical Reference(s):

PSTG/SATG AMP-TBD00 rev 8 APDX B
EOP 5.8.1, RPV Depressurization Systems (Table 2), Rev. 40

Proposed references to be provided to applicants during examination: None

Learning Objective:

6. Identify any EOP support procedure addressed in Flowchart 2A and apply any associated special operating instructions or cautions.

7. Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION / STEAM COOLING, determine required actions.

Question Source:

Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	1
K/A #	2.1.4
Importance Rating	3.8

2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Question: 94

Which one of the following completes the statements below regarding SRO License reactivation requirements?

A minimum of ____(1)____ 12-hour shifts under instruction is required to reactivate.

A "Special Prescription Respirator Glasses Verification" is required IAW ____(2)____?

A. (1) four
(2) NTP8.1 (Administration of Licensed Operator Medical Examination Program)

B. (1) four
(2) 2.0.7 (Licensed Operator Active/Reactivation/Medical Status Maintenance Program)

C. (1) five
(2) NTP8.1 (Administration of Licensed Operator Medical Examination Program)

D. (1) five
(2) 2.0.7 (Licensed Operator Active/Reactivation/Medical Status Maintenance Program)

Answer:

B. (1) four
(2) 2.0.7 (Licensed Operator Active/Reactivation/Medical Status Maintenance Program)

Explanation:
Requires knowledge that four 12 hour shift watches are required to support license reactivation. It is plausible to choose five since this is required to support license maintenance (five 12 hour shifts under instruction in the current or previous quarter would satisfy the required hours). The Special Prescription Respirator Glasses Verification is contained on the SRO On-Shift Time & Reactivation Attachments in procedure 2.0.7. If a change in medical condition were to occur, Procedure 2.0.7 directs licensee to complete Attachment 2 of Procedure NTP8.1. It is plausible to choose procedure NTP8.1 due being titled "Medical Status Maintenance Program".

Distracters:

A. This answer is incorrect because NTP8.1 does not contain the Special Prescription Respirator Glasses Verification. This choice is plausible due to this procedure being required to support change in licensed operator medical status. The candidate who correctly identifies the number of watches required to reactivate and confuses Respirator Glasses Verification with a change in license medical status would select this answer.

C. This answer is incorrect due to the number of shifts required and NTP8.1 not containing the Special Prescription Respirator Glasses Verification. This choice is plausible due 5 shifts being required for initial license activation and quarterly license maintenance and this procedure being required to support change in licensed operator medical status. The candidate who confuses the number of watches required to activate/maintain vs. reactivate and confuses Respirator Glasses Verification with a change in license medical status would select this answer.

D. This answer is incorrect due to the number of shifts required. This choice is plausible due 5 shifts being required for initial license activation. The candidate who confuses the number of watches required to activate/maintain vs. reactivate and correctly identifies the procedure requiring Special Prescription Respirator Glasses Verification would select this answer.

Technical Reference(s):
Procedure 2.0.7 (Licensed Operator Active/Reactivation/Medical Status Maintenance Program) Rev. 09
Procedure NTP8.1 (Administration of Licensed Operator Medical Examination Program), Rev. 17

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT00705130010100
Given a set of plant conditions, recognize non-compliance with a Chapter 5.0 Requirement.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10
55.43.2

LOD 4

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Requires knowledge of NRC license maintenance requirements and selection of procedure requiring Special Prescription Respirator Glasses Verification.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	1
K/A #	2.1.41
Importance Rating	3.7

2.1.41 Knowledge of the refueling process.

Question: 95

Which activity requires Refuel Floor Supervisor permission during refueling operations in Mode 5?

A. Allowing under vessel access.

B. Allowing access to the refuel floor.

C. Re-commencing fuel handling operations.

D. Using greater than 50 gallons of demineralized water on the refuel floor.

Answer:

C. Re-commencing fuel handling operations.

Explanation:
Requires knowledge of Refuel Floor SRO responsibilities during refueling operations. Refuel Floor Supervisor permission is required to re-commence fuel handling operations IAW Attachment 4 (Reset Checklist) which shall be used each time the normal fuel handling process is stopped/interrupted. This includes, but is not limited to, Shift Turnover, Fuel Mover/Spotter mid-shift role change, or following a distraction which interrupts the normal fuel handling process flow.

Distracters:

A. This answer is incorrect because Refuel Floor Supervisor permission is not required to allow access to the under vessel area. This answer is plausible because under vessel area gets posted to prohibit access without Shift Manager's permission. The candidate who confuses access permission authority would select this choice.

B. This answer is incorrect because Refuel Floor Supervisor permission is not required to allow access to the refuel floor. This choice is plausible due to the Refuel floor SRO permission is required to access the refueling area. The candidate who confuses refuel floor with refuel area would choose this answer.

D. This answer is incorrect because Refuel Floor Supervisor permission is not required to use greater than 50 gallons of demineralized water on the refuel floor. This choice is plausible due to the Refuel floor SRO is required to brief available refueling floor personnel on limiting demineralized water usage and requirement to notify Control Room if using > 50 gallons demineralized water each shift. The candidate who confuses briefing vs. giving permission would choose this answer.

Technical Reference(s):
Procedure 2.1.20.3, RPV Refueling Preparation (Wet Lift of Dryer and Separator), Rev. 45
Procedure 10.25 (Refueling - Core Unload, Reload, and Shuffle), Rev. 59
Procedure 2.2.31 (Fuel Handling - Refueling Platform), Rev. 47

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT0231002001160A Identify the administrative duties and responsibilities of the each of the following: Refueling Floor Supervisor

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.2
55.41.10
55.43.6
55.45.13

LOD 3

10 CFR 55.43(b)(7) - Fuel handling facilities and procedures.
Requires knowledge of Refuel floor SRO responsibilities.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	2
K/A #	2.2.11
Importance Rating	3.3

2.2.11 Knowledge of the process for controlling temporary design changes.

Question: 96

The plant is operating at power in Mode 1.

What is the MAXIMUM time a Temporary Alteration In Support of Maintenance (TASM) can be installed on plant equipment WITHOUT a 10CFR50.59 Review?

A. 30 days

B. 60 days

C. 90 days

D. 120 days

ANSWER:

C. 90 days

Explanation:
A temporary alteration is necessary to support maintenance if it makes the maintenance activity easier, or the maintenance activity has been planned to allow prompt restoration. TASMs have regulatory considerations specific to duration under 10CFR50.59. Engineering Procedure 3.4.4, Temporary Configuration Change, Attachment 7, Step 4.1.1 and 4.1.2 describe the requirements for a 10CFR50.59 review prior to installation if it is expected to be in place > 90 days, or if after installation, it is going to be installed > 90 days.

Distracters:

A. This answer is incorrect because the time listed is not the maximum time a TASM can be installed without a 10CFR50.59 Review being performed. The maximum time is 90 days as allowed by federal regulations. This answer is plausible because some non-emergency events require NRC notification reports and the candidate may recall the 30 days without tying it to the TASM requirements.

B. This answer is incorrect because the time listed is not the maximum time a TASM can be installed without a 10CFR50.59 Review being performed. The maximum time is 90 days as allowed by federal regulations. This answer is plausible because configuration change affected documents must be processed within 60 days of CED installation and the candidate may remember that time frame.

D. This answer is incorrect because the time listed exceeds the maximum time a TASM can be installed without a 10CFR50.59 Review being performed. The maximum time is 90 days as allowed by federal regulations. This answer is plausible because there is a time limit of 120 days in Tech Specs for time shutdown which requires scram timing on startup prior to 40% power. The candidate may recall the 120 days but be unsure of its source.

Technical Reference(s):
Procedure 3.4.4 (Temporary Configuration Change), Rev. 15.

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT0320109A0A010A
Procedure 3.4.4, Temporary Configuration Change - Discuss the following as described in Engineering Procedure 3.4, Configuration Change Control: Temporary Configuration Changes (TCCs)

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10
55.43.3
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (3) Facility licensee procedure required to obtain authority for design and operating changes in the facility.
Requires knowledge of Administrative processes for temporary modifications/configuration changes.
SRO Task: 200001W0303 Approve Installation of a Plant Temporary Configuration Change (TCC) Order. The SRO is responsible for knowledge of the TCC process.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	2
K/A #	2.2.19
Importance Rating	3.4

2.2.19 Knowledge of maintenance work order requirements.

Question: 97

Which one of the following identifies items required to be addressed by an SRO for a Work Order Standard Plant Impact Statement IAW 0-CNS-WM-102 (Work Implementation and Closeout)?

A. Identification of power reduction requirements and any special plant conditions.

B. Identification of power reduction requirements and determination of required tools.

C. Verification of required associated parts availability and any special plant conditions.

D. Verification of required associated parts availability and determination of required tools.

Answer:
A. Identification of power reduction requirements and any special plant conditions.

Explanation: Requires knowledge of SRO/FIN SRO/WCCA/WCC Supervisor impact review of maintenance work orders. The Standard Plant Impact Statement addresses the following items:

1. Does the work activity affect SSCs identified in TSs?
2. Will the maintenance activity require the equipment to be declared inoperable/unavailable?
3. Is a power reduction required?
4. Any other special plant conditions required to perform this work activity?
5. Is there a potential to affect the Operability of systems, structures, or components required to be operable?
6. Does the maintenance activity create the potential for inadvertent actuations (RPS trip, turbine trip, ECCS actuation, SDC isolation, system discharge valve closure)?
7. Does the maintenance activity have an actual or potential reactivity impact? (CRD, recirc flow, feed flow, feed temperature, etc.)?
8. Does the work activity increase the potential for loss of off-site power?
9. Will single valve isolation be required?
10. Are contingency plans or compensatory actions necessary?

Distracters:

B. This answer is incorrect due to determination of required tools not being part of the impact review by operations. This answer is plausible due to needing tools to perform the maintenance, but is not required to be verified by the operations organization. The candidate who correctly recognizes power reduction and believes tool verification is also verified during the operations impact review would select this answer.

C. This answer is incorrect due to verification of required parts not being part of the impact review by operations. This answer is plausible due to needing parts to perform the maintenance, but is not required to be verified by the operations organization. The candidate who incorrectly believes parts verification and correctly identifies special plant conditions required during the operations impact review would select this answer.

D. This answer is incorrect due to determination of required parts & tools not being part of the impact review by operations. This answer is plausible due to needing parts & tools to perform the maintenance, but is not required to be verified by the operations organization. The candidate who incorrectly believes parts & tool verification is required during the operations impact review would select this answer.

Technical Reference(s):
Procedure CNS-WM-102, Work Implementation and Closeout, Rev.01.

Proposed references to be provided to applicants during examination: None

Learning Objective: SKL0110101001290A, 0.40, Work Control Program, Discuss the following as described in Administrative Procedure 0.40, Work Control Program: 1) Precautions and limitations 2) Minor maintenance 3) Emergency MWR processing.

Question Source:
Bank #
Modified Bank #
New X

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.13

LOD 3

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Knowledge of administrative procedures that specify implementation of plant normal procedures.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	3
K/A #	2.3.6
Importance Rating	3.8

2.3.6 Ability to approve release permits.

Question: 98

Waste Sample Tank A is required to be discharged while shutdown.

What are the MINIMUM CW flow requirements, and the MINIMUM tank recirculation times which allow approval of the release by the Shift Manager IAW Procedure 8.8.11 (Liquid Radioactive Waste Discharge Authorization)?

A. 1 CW pump operating with De-Icing in service AND
Tank has been recirculated for a MINIMUM of 30 minutes prior to being sampled.

B. 1 CW pump operating with De-Icing in service AND
Tank has been recirculated for a MINIMUM 2 hours prior to being sampled.

C. 1 CW pump operating with De-Icing NOT in service AND
Tank has been recirculated for a MINIMUM 30 minutes prior to being sampled.

D. 1 CW pump operating with De-Icing NOT in service AND
Tank has been recirculated for a MINIMUM 2 hours prior to being sampled.

Answer:
D. 1 CW pump operating with De-Icing NOT in service AND
Tank has been recirculated for a MINIMUM 2 hours prior to being sampled.

Explanation:
The following items are verified by the Shift Manager prior to approving a liquid RW discharge:

- Tank recirculation time > 2 hours prior to being sampled
- Proper dilution flow (159,000 gpm)
- Tank volume sampled matches volume to be released
- DISCHARGE IN PROGRESS Tags Installed On Running CW Pumps.

Distracters:

A. This answer is incorrect due to 1 CW pump operating with De-Icing in service not providing sufficient dillution flow (118,800 gpm vs. 159,000 gpm) and tank recirculation not meeting the 2 hour procedural requirement. This answer is plausible due to 1 CW pump operating with De-icing NOT in service would provide sufficient dilution flow and 30 minutes is the normal tank recirculation time prior to sampling to support transfer vs. discharge. The candidate who confuses De-Icing in service and tank sample recirc time for transfer would select this answer.

B. This answer is incorrect due to 1 CW pump operating with De-Icing in service not providing sufficient dillution flow (118,800 gpm vs. 159,000 gpm). This answer is plausible due to 1 CW pump operating with De-icing NOT in service would provide sufficient dilution flow. The candidate who confuses De-Icing in service and correctly recognizes tank sample recirc time for discharge would select this answer.

C. This answer is incorrect due to tank recirculation not meeting the 2 hour procedural requirement. This answer is plausible due to 30 minutes being the normal tank recirculation time prior to sampling to support transfer vs. discharge. The candidate who correctly recognizes proper dilution flow and confuses De-Icing in service and tank sample recirc time for transfer would select this answer.

Technical Reference(s):
Procedure 8.8.11 Liquid Radioactive Waste Discharge Authorization, Rev. 32.
Procedure 2.5.1.6 (Radwaste Low Conductivity Liquid Waste Sample Tank Fluid Transfer), Rev. 43

Proposed references to be provided to applicants during examination: None

Learning Objective:

INT0320115B0B0200 State who, by title, is required to grant permission to perform a liquid discharge with the liquid radwaste monitor inoperable.

Question Source:
Bank # 2304
Modified Bank #
New

Question Cognitive Level: Memory or Fundamental Knowledge x
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.13
55.43.4
55.45.10

LOD 3

SRO Only - 10CFR55.43 b (4) Radiation hazard that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
SROs are required to have knowledge of the process for liquid release approval.

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	4
K/A #	2.4.29
Importance Rating	4.4

2.4.29 Knowledge of the Emergency Plan.

Question: 99

Given the following:

- At 1200 the threshold for a NOUE is exceeded.
- At 1210 the Emergency Director declared classification of a NOUE.
- At 1215 the threshold for an ALERT is exceeded.
- At 1220 the Emergency Director declared classification of an ALERT.

Which of the following identifies the LATEST time that State/Local agency notifications of the NOUE classification is required to be performed AND IAW EPIP 5.7.6 (Notification)?

A. 1215

B. 1225

C. 1230

D. 1235

Answer:

B. 1225

Explanation:
Requires knowledge of E-Plan notification requirements. Initial notification to State/local agencies of E-plan classification is required to be performed within 15 minutes of emergency declaration. There is a common mis-application of start times when staggering EALs are met. The SRO must keep separate times running for each EAL as it is entered and not reset an earlier time due to subsequent EALs. This is the responsibility of the Emergency Director (SRO) until relieved by another Emergency Director.

Distracters:

A. This answer is incorrect due to not being the latest time requiring notification. This answer is plausible due to being 15 minutes from exceeding an EAL threshold. The candidate who cannot differentiate between the time to notify State/Local from exceeding EAL threshold vs. Declaration would choose this answer.

C. This answer is incorrect due exceeding the latest time requiring notification. This answer is plausible due to being 15 hour from exceeding an Alert EAL threshold. The candidate who cannot differentiate between the time to notify State/Local from Alert EAL threshold vs. NOUE Declaration would choose this answer.

D. This answer is incorrect due exceeding the latest time requiring notification. This answer is plausible due to being 15 minutes from the ALERT declaration. The candidate who cannot differentiate between the time to notify State/Local from Alert EAL Declaration vs. NOUE Declaration would choose this answer.

Technical Reference(s):
EPIP 5.7.2 (Emergency Director EPIP), Rev. 32
Emergency Procedure 5.7.6 (Notification), Rev 24.

Proposed references to be provided to applicants during examination: None

Learning Objective:
GEN0030401B0B030B Emergency Notifications and Communications Systems: State the time requirements for initial and/or follow-up notifications to offsite agencies.

Question Source:
Bank #
Modified Bank # 6082
New

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.13

LOD 2

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
The SRO is responsible for task 344022O0303, Direct Emergency Response as Emergency Director (Emergency Plan)

QUESTION: 88 6082 (1 point(s))

Following Classification of an emergency event, what are the time interval guidelines for notifying the State **AND** Local Authorities of the emergency classification?

Notification must be made . . .

a. immediately.

b. within 15 minutes.

c. within 30 minutes.

d. within 60 minutes.

ANSWER: 88 6082

b. within 15 minutes.

EXPLANATION OF ANSWER: b. Correct. a,c,d. Incorrect.

5.7.6, step 2.1, rev 44

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	4
K/A #	2.4.5
Importance Rating	4.3

2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Question: 100

Which one of the following identifies the type of procedure that takes precedence if a procedure conflict arises during plant operations AND the procedure which provides this guidance?

A. Alarm over Abnormal IAW Procedure 2.0.1.2 (Ops Policy Procedure).

B. Emergency Operating over System Operating IAW Procedure 2.0.1.2 (Ops Policy Procedure).

C. Alarm over Abnormal IAW Procedure 0-EN-HU-106 (Procedure and Work Instruction Use and Adherence).

D. Emergency Operating over System Operating IAW Procedure 0-EN-HU-106 (Procedure and Work Instruction Use and Adherence).

Answer:

B. Emergency Operating over System Operating IAW Procedure 2.0.1.2 (Ops Policy Procedure).

Explanation:
Alarm/Abnormal/Emergency/System Operating Procedures/Instrument Operating Procedures may be carried out concurrently with an EOP. In the event that conflicting actions are directed by procedures, the EOP actions shall take precedence. EOPs/SAMGs are Operations highest tier procedure. If an explicit operation is directed by EOPs per a 5.8 EOP Support Procedure, then transition shall be made from the Alarm/Abnormal/Emergency/System Operating/Instrument Operating Procedures (including hard cards) to the 5.8 Procedure to perform or continue performing that operation. The SRO selects procedure guidance based on procedure heirarchy.

Distracters:

A. This answer is incorrect due to APs taking precedence over Alarm procedures. This answer is plausible if hierarchy is confused. The candidate who confuses hierarchy and correctly identifies the procedure which provides hierarchy guidance would choose this answer.

C. This answer is incorrect due to APs taking precedence over Alarm procedures and 0-EN-HU-106 not providing hierarchy guidance. This answer is plausible if hierarchy is confused and it is reasonable to assume the hierarchy guidance is contained within 0-EN-HU-106. The candidate who confuses hierarchy and assumes the hierarchy guidance is contained within 0-EN-HU-106 would choose this answer.

D. This answer is incorrect due to 0-EN-HU-106 not providing hierarchy guidance. This answer is plausible due to being reasonable to assume the hierarchy guidance is contained within 0-EN-HU-106. The candidate who correctly identifies the hierarchy and assumes the hierarchy guidance is contained within 0-EN-HU-106 would choose this answer.

Technical Reference(s):
Procedure 2.0.1.2, Operations Procedure Policy, Rev 44
Procedure 0-EN-HU-106 (Procedure and Work Instruction Use and Adherence), Rev. 2C0

Proposed references to be provided to applicants during examination: None

Learning Objective:
INT0320101R2 Describe the hierarchy between the Emergency Operating Procedures, Abnormal Procedures, and Emergency Procedures, including which guidance takes precedence.

Question Source:
Bank #
Modified Bank # 3932
New

Question History: Last NRC Exam 2011

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10
55.43.5
55.45.3

LOD 2

SRO Only - 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

QUESTION: 43 3932 (1 point(s))

Should a procedure conflict arise during plant operations, which of the below type procedures would take precedence?

a. Alarm.

b. Abnormal.

c. System Operating.

d. Emergency Operating.

ANSWER: 43 3932

d. Emergency Operating.

REFERENCE: PROCEDURE 2.0.1.2 states:

"Alarm/Abnormal/Emergency/System Operating Procedures/Instrument Operating Procedures may be carried out concurrently with an EOP. In the event that conflicting actions are directed by procedures, the EOP actions shall take precedence. EOPs/SAMGs are Operations highest tier procedure. If an explicit operation is directed by EOPs per a 5.8 EOP Support Procedure, then transition shall be made from the Alarm/Abnormal/Emergency/System Operating/Instrument Operating Procedures (including hard cards) to the 5.8 Procedure to perform or continue performing that operation."

RO References

NOTE 3

Torus overpressure is sum of torus pressure and hydrostatic head above suction strainer

Torus pressure (psig)			________
Hydrostatic head (psig)			
PC water level (ft.)	________		
Strainer level (ft.)	-4		
0.43 x	________	=	+ ________
Torus overpressure (psig)			________

DRYWELL TEMPERATURE

1

DW/T-1

Monitor average drywell temperature, EOP 5.8.10, and control below 150°F using available drywell cooling

DW/T-2

WHEN

average drywell temperature cannot be maintained below 150°F

DW/T-3

Operate all available drywell cooling (defeat isolation interlocks if necessary)

DW/T-4

BEFORE

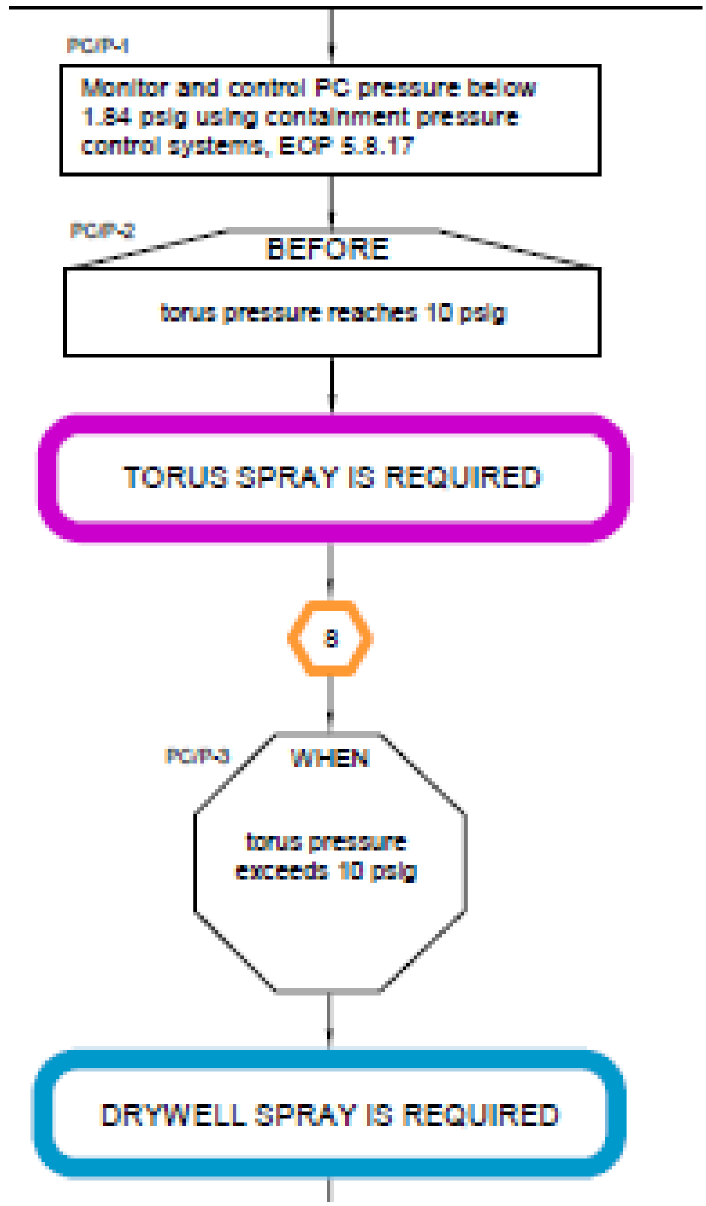
average drywell temperature reaches 280°F

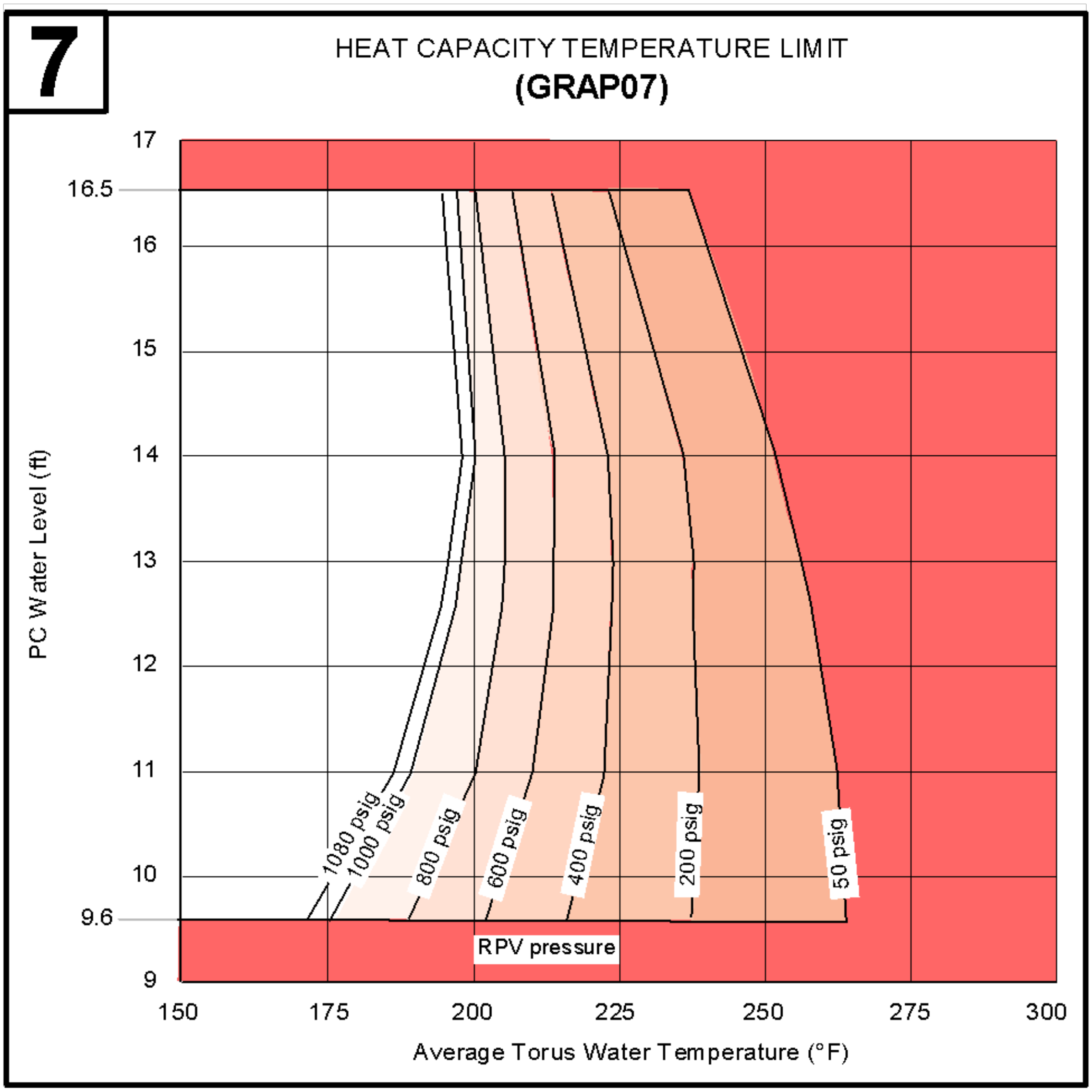
concurrently

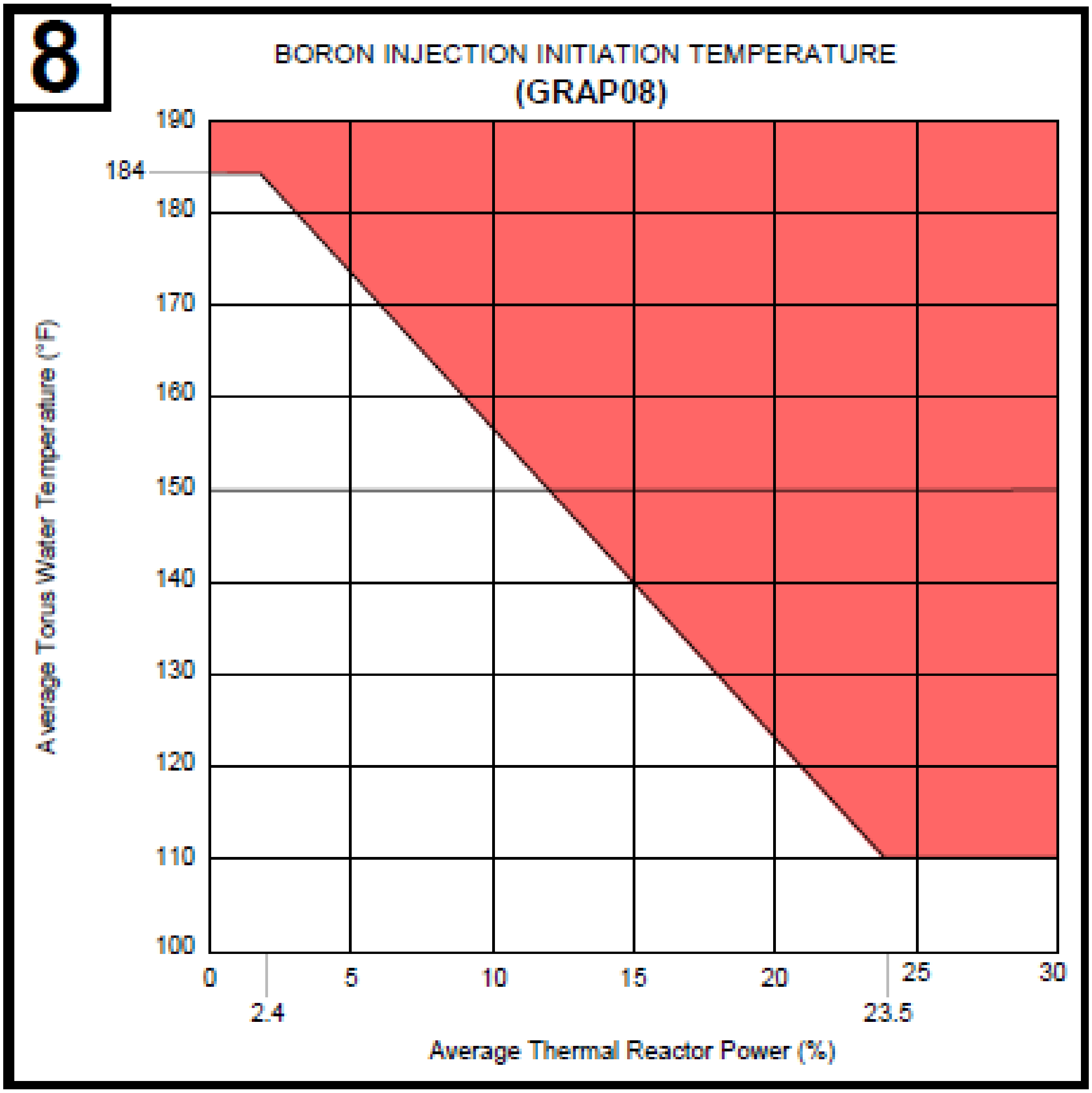
1A, 1

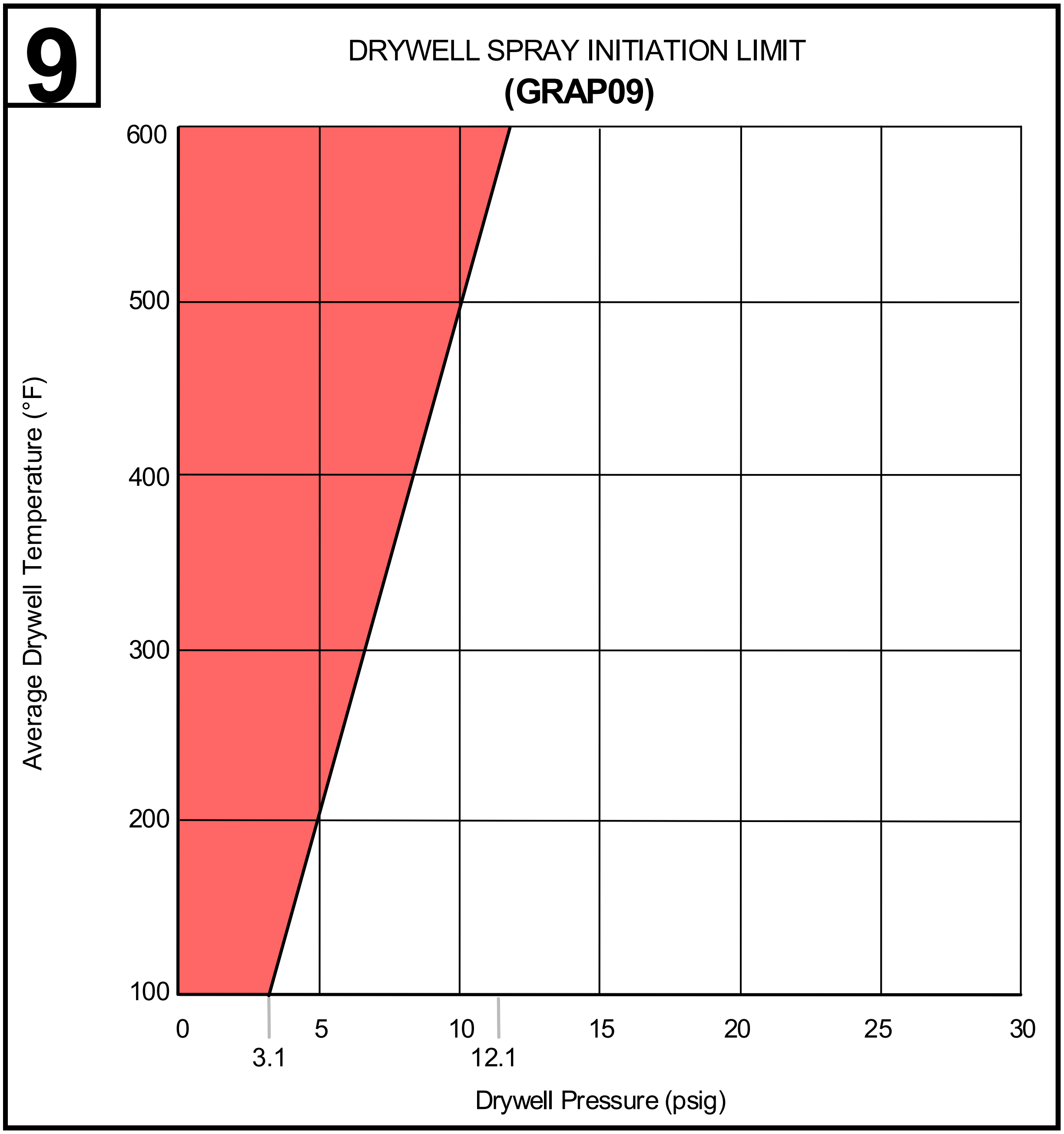
DRYWELL SPRAY IS REQUIRED

PC PRESSURE


PC/P-1
Monitor and control PC pressure below 1.84 psig using containment pressure control systems, EOP 5.8.17
PC/P-2
BEFORE
torus pressure reaches 10 psig
TORUS SPRAY IS REQUIRED
8
PC/P-3
WHEN
torus pressure exceeds 10 psig
DRYWELL SPRAY IS REQUIRED


7
HEAT CAPACITY TEMPERATURE LIMIT
(GRAP07)
PC Water Level (ft)
17
16.5
16
15
14
13
12
11
10
9.6
9
1080 psig
1000 psig
800 psig
600 psig
400 psig
200 psig
50 psig
RPV pressure
150
175
200
225
250
275
300
Average Torus Water Temperature (°F)

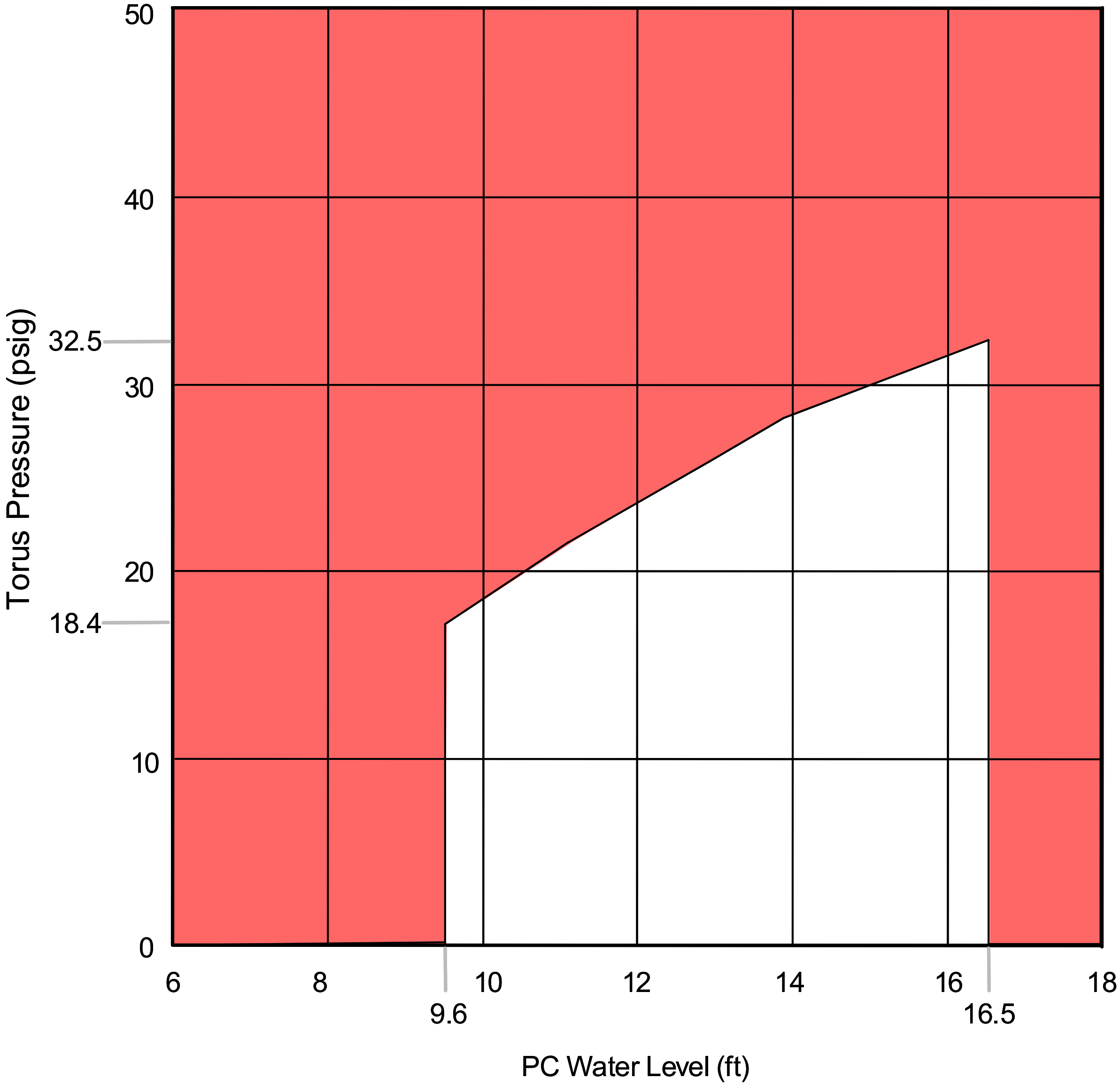

8
BORON INJECTION INITIATION TEMPERATURE
(GRAP08)
Average Torus Water Temperature (°F)
190
184
180
170
160
150
140
130
120
110
100
0
2.4
5
10
15
20
23.5
25
30
Average Thermal Reactor Power (%)

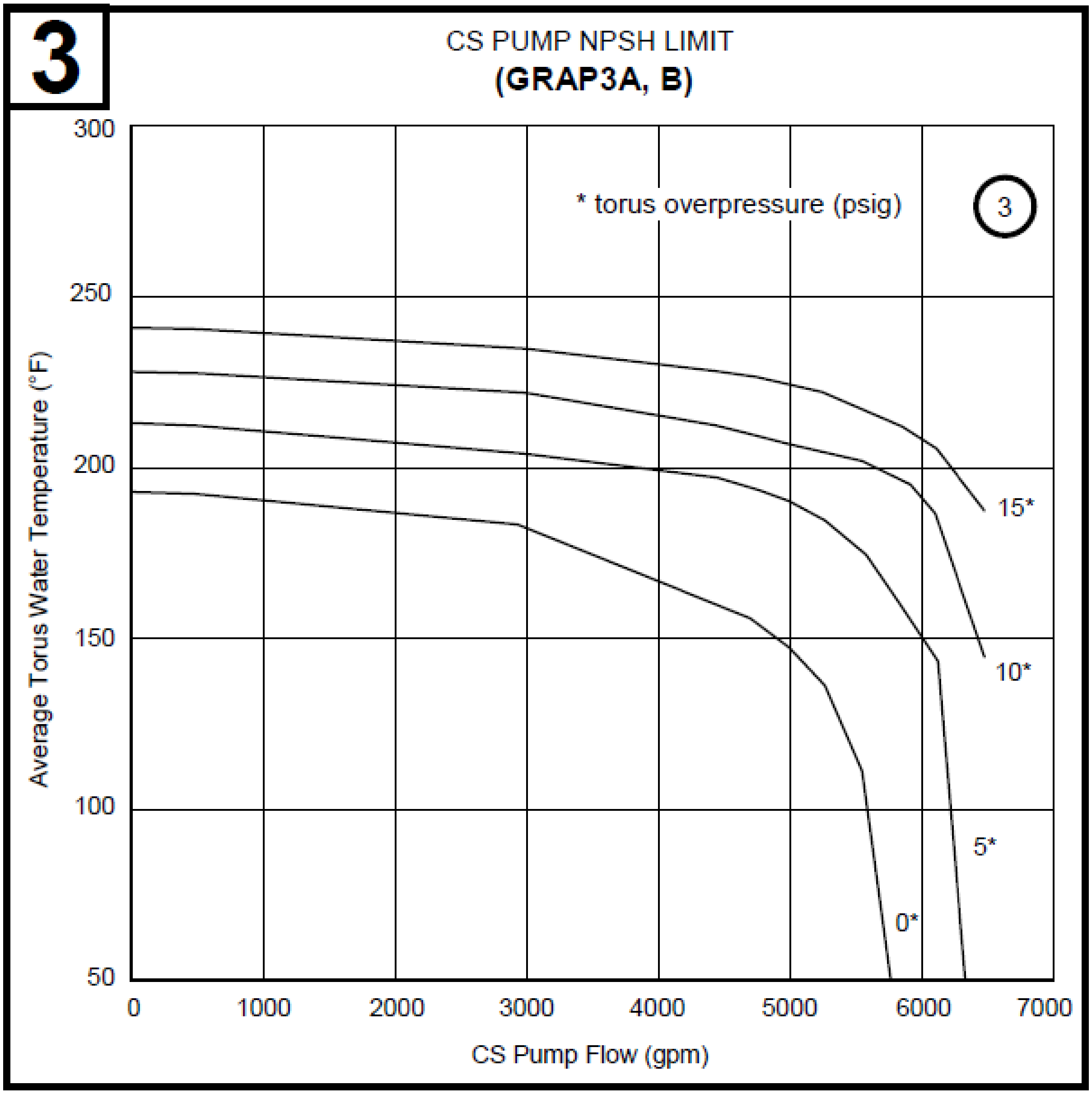

9
DRYWELL SPRAY INITIATION LIMIT
(GRAP09)
600
500
400
300
200
100
Average Drywell Temperature (°F)
0
5
10
15
20
25
30
3.1
12.1
Drywell Pressure (psig)

10

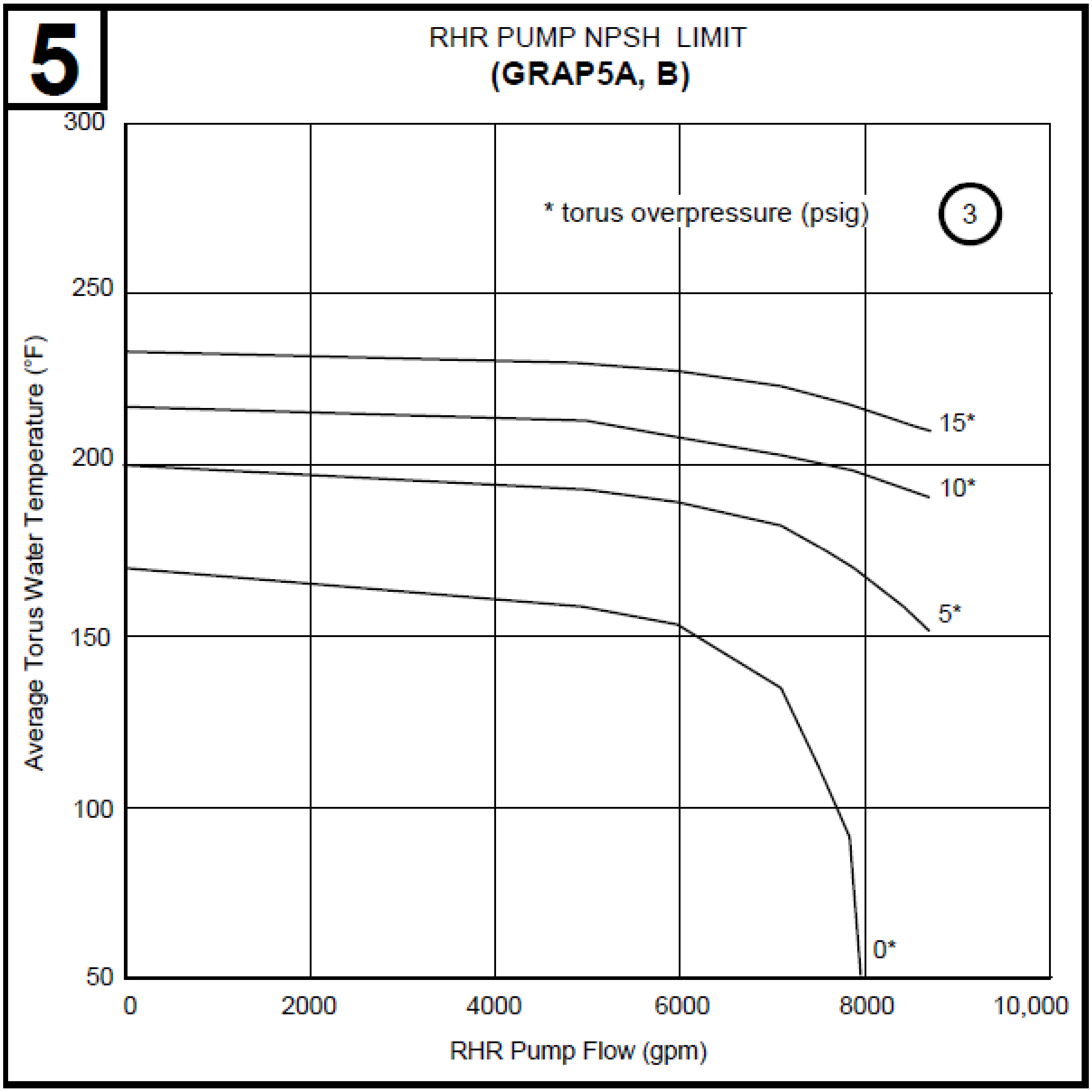
PRESSURE SUPPRESSION PRESSURE

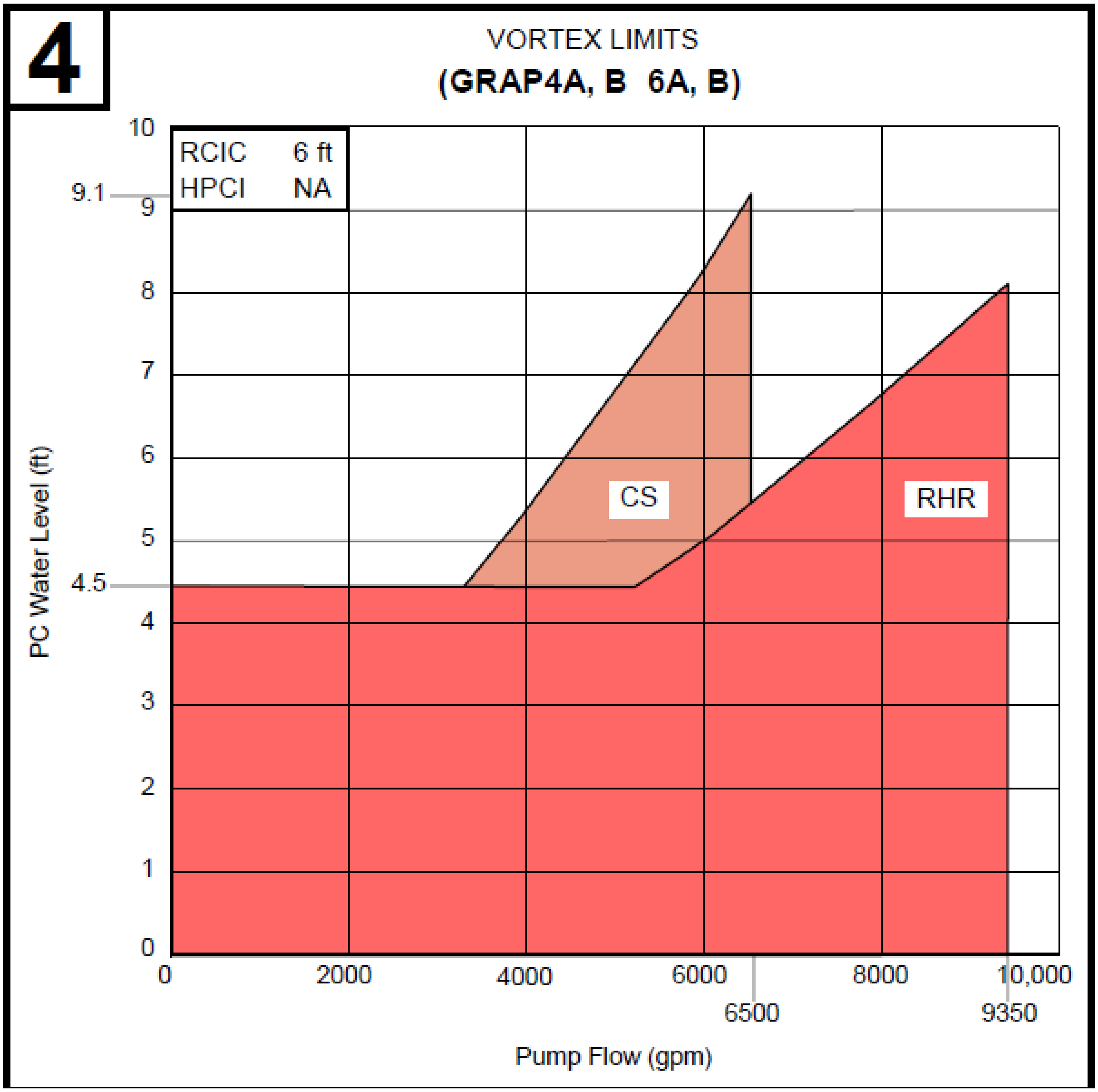
(GRAP10)

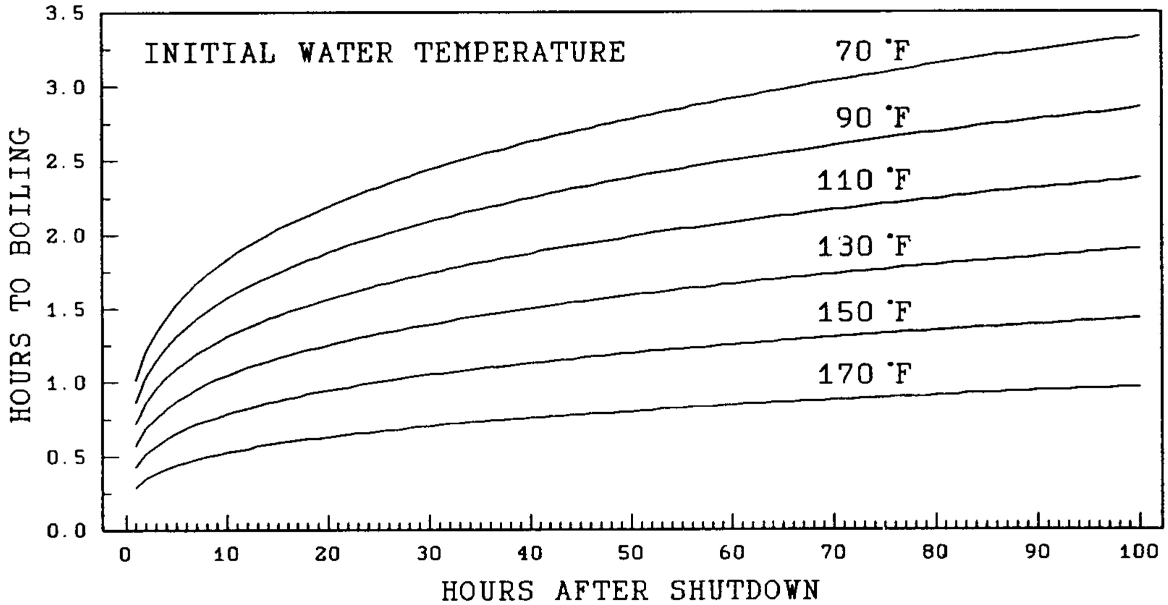




3
CS PUMP NPSH LIMIT
(GRAP3A, B)
* torus overpressure (psig)
3
15*
10*
5*
0*
300
250
200
150
100
50
0
1000
2000
3000
4000
5000
6000
7000
Average Torus Water Temperature (°F)
CS Pump Flow (gpm)


5
RHR PUMP NPSH LIMIT
(GRAP5A, B)
* torus overpressure (psig)
3
15*
10*
5*
0*
300
250
200
150
100
50
Average Torus Water Temperature (°F)
0
2000
4000
6000
8000
10,000
RHR Pump Flow (gpm)


4
VORTEX LIMITS
(GRAP4A, B 6A, B)
RCIC 6 ft
HPCI NA
CS
RHR
PC Water Level (ft)
10
9.1
9
8
7
6
5
4.5
4
3
2
1
0
0
2000
4000
6000
6500
8000
10,000
9350
Pump Flow (gpm)



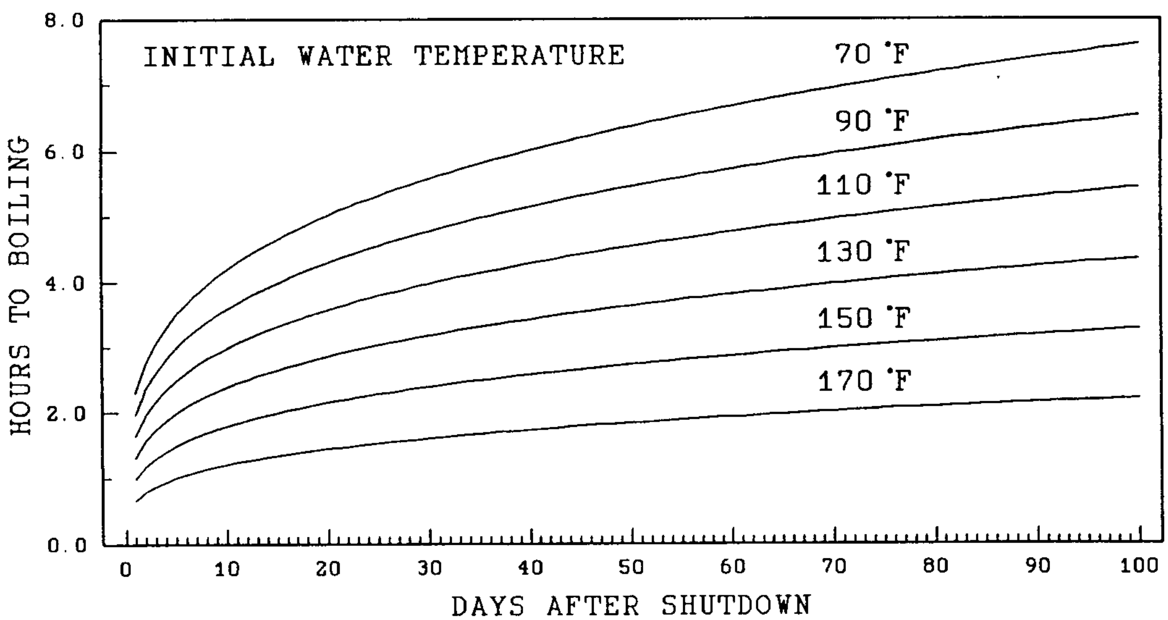
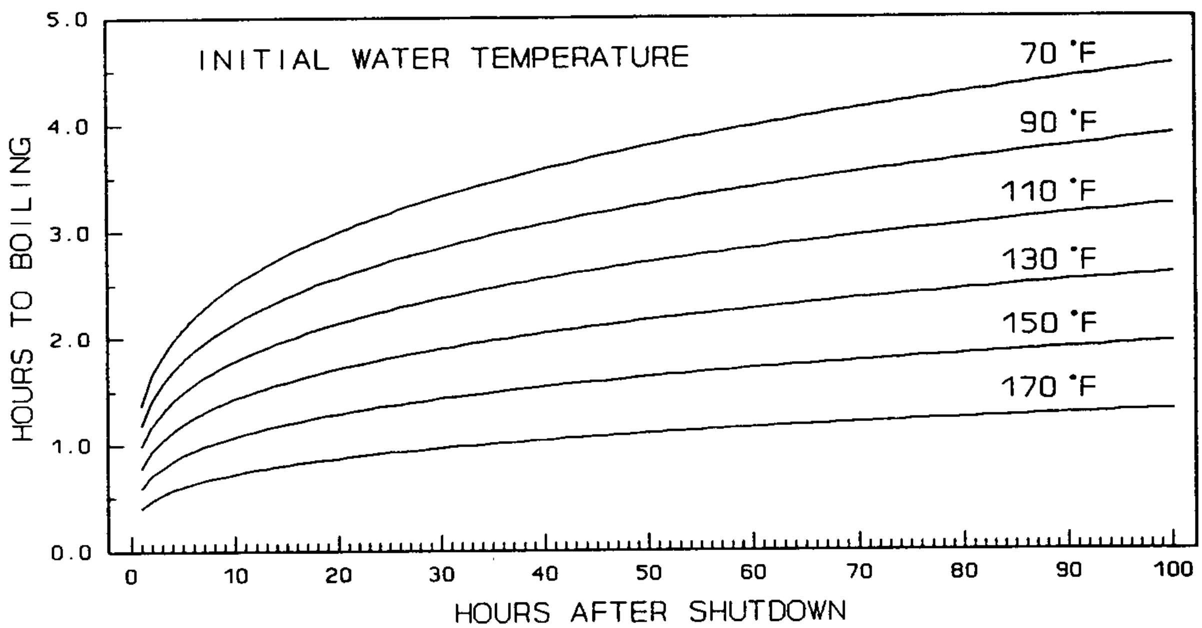


Figure 1 - TIME TO BOILING - WATER LEVEL AT HIGH LEVEL TRIP



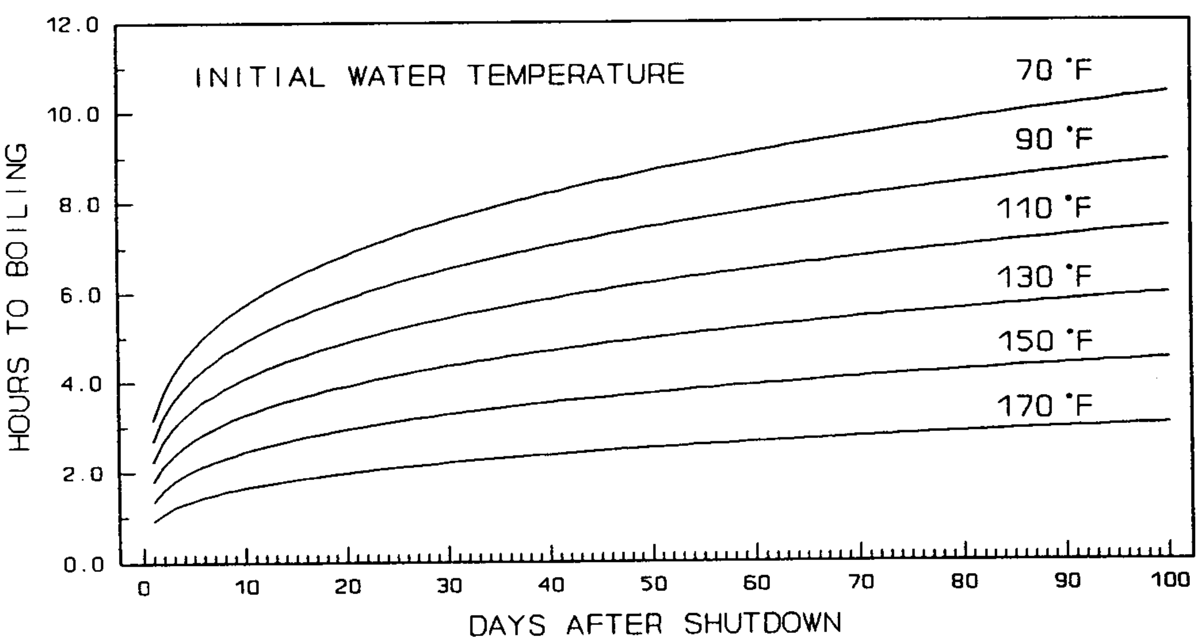
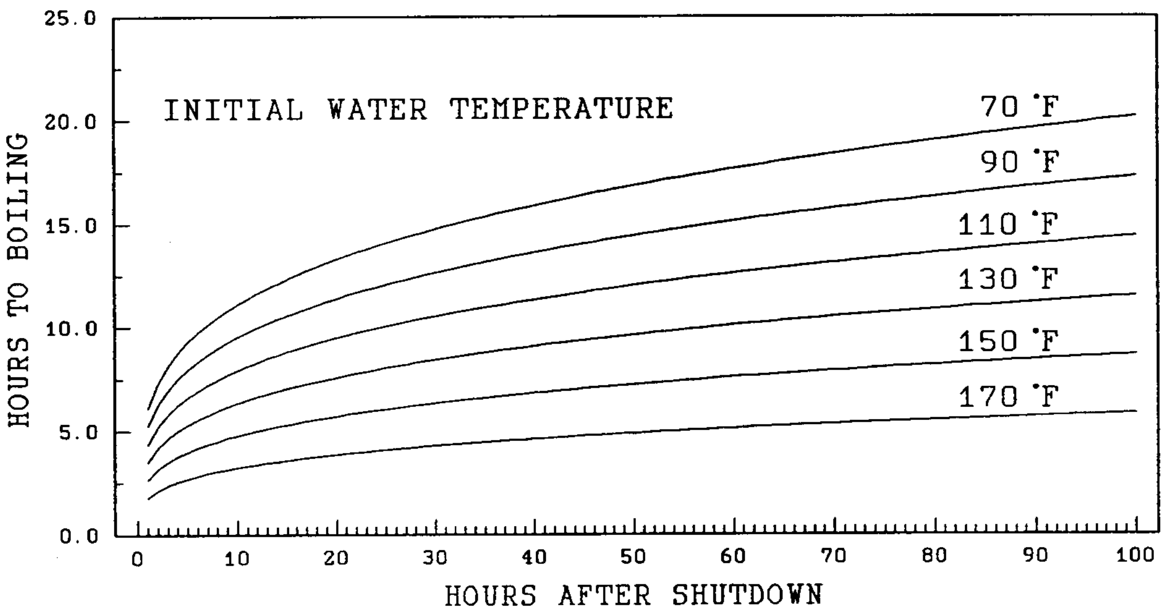


Figure 2 - TIME TO BOILING - WATER LEVEL AT FLANGE



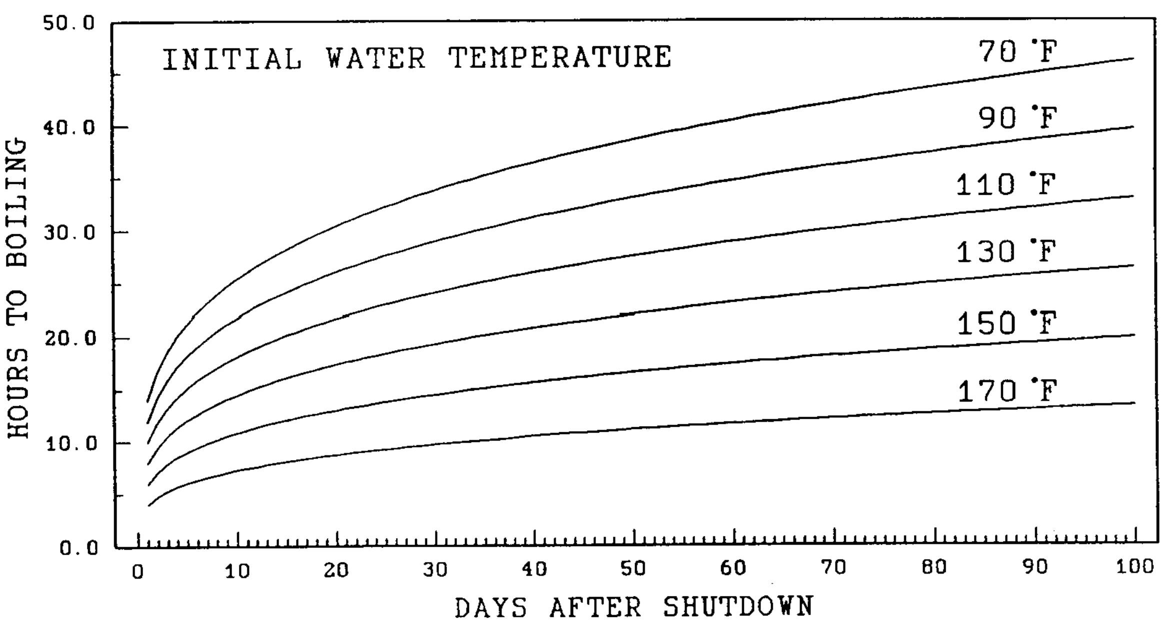
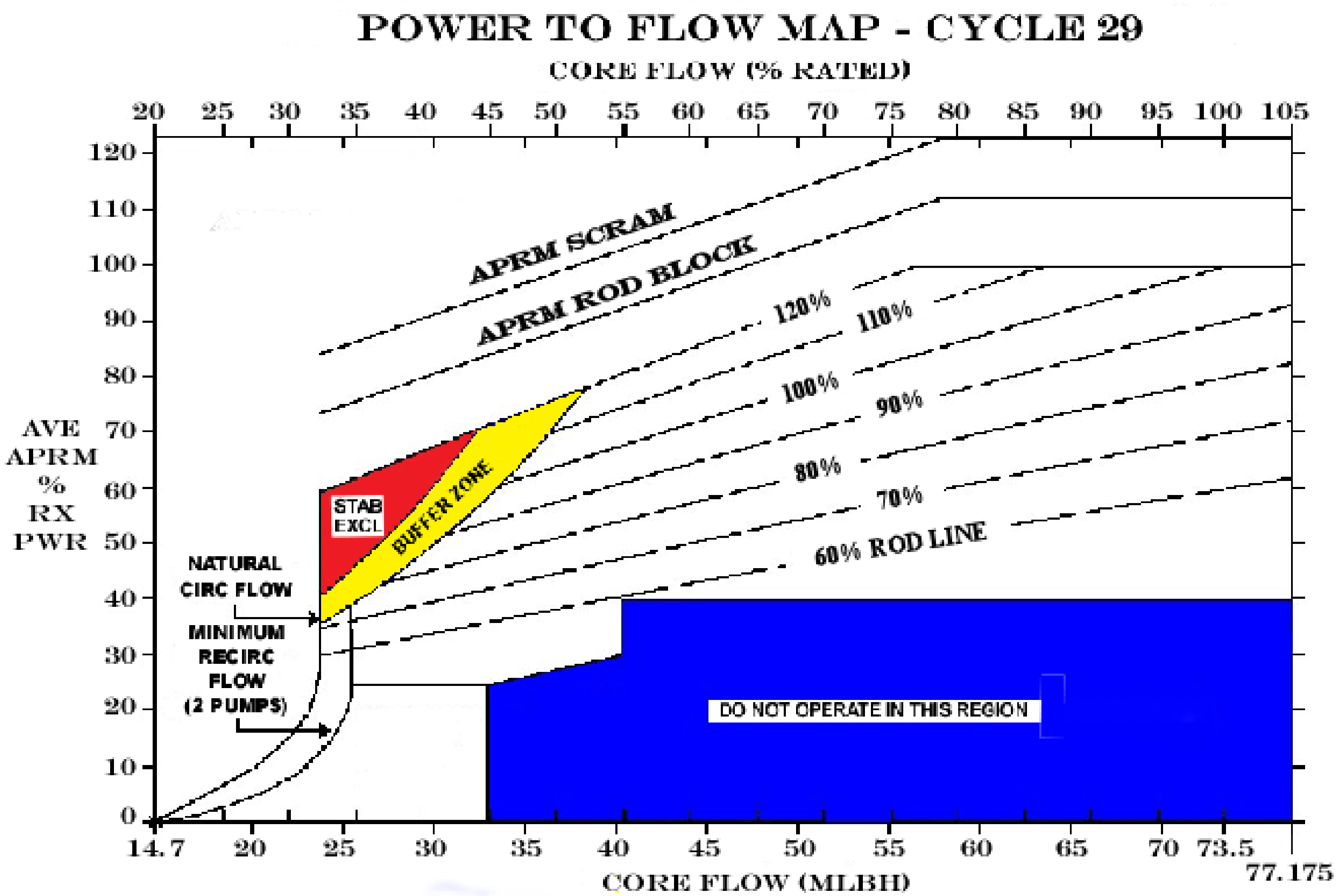


Figure 3 - TIME TO BOILING - WATER TO LEVEL FLOODED TO 1001'



NOTE – The Maximum Effective Load Line Limit (MELLL) is 118.9% RTP (line not shown on PMIS screen).

SRO References

T 3.11 FIRE PROTECTION SYSTEMS

T 3.11.2 Fire Suppression Water System

TLCO 3.11.2 The Fire Suppression Water System shall be OPERABLE with:

a. Two OPERABLE fire pumps with their discharge aligned to the fire suppression header.

b. An OPERABLE flow path capable of taking suction from either of two 500,000 gallon water storage tanks and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant valves and the front valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required fire pump inoperable.	A.1 Restore fire pump to OPERABLE status.	7 days

(continued)

INFORMATION ONLY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. The Fire Suppression Water System inoperable for reasons other than Condition A.	B.1 Establish a backup fire suppression water system.	24 hours
	AND	
	B.2 Restore Fire Suppression Water System to OPERABLE status.	7 days

INFORMATION ONLY

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. <u>OR</u> A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. <u>OR</u> B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 29.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
10. Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.11 SR 3.3.1.1.13	NA
	5[c]	1	H	SR 3.3.1.1.11 SR 3.3.1.1.13	NA

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The Limiting Trip Setpoint and the methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Requirements Manual.

(c) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify reactor steam dome pressure is	12 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources — Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable to DGs

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one division concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
		(continued)

INFORMATION ONLY

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Restore offsite circuit to OPERABLE status.	7 days AND 14 days from discovery of failure to meet LCO
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	B.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	<u>OR</u>	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	<u>AND</u>	
	B.4 Restore DG to OPERABLE status.	7 days <u>AND</u> 14 days from discovery of failure to meet LCO
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	<u>AND</u>	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One offsite circuit inoperable. AND One DG inoperable.	------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems — Operating," when Condition D is entered with no AC power source to either division. ----------------------------	
	D.1 Restore offsite circuit to OPERABLE status.	24 hours
	OR	
	D.2 Restore DG to OPERABLE status.	24 hours
E. Two DGs inoperable.	E.1 Restore one DG to OPERABLE status.	2 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3.	12 hours
	AND	
	F.2 Be in MODE 4.	36 hours
G. Three or more required AC sources inoperable.	G.1 Enter LCO 3.0.3.	Immediately

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

A — Abnorm. Rad Release / Rad Effluent

1 — Offsite Rad Conditions

GENERAL EMERGENCY

AG1.1 1 2 3 4 5 DEF

Any valid gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)

AG1.2 1 2 3 4 5 DEF

Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary

AG1.3 1 2 3 4 5 DEF

Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)

OR

Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary

SITE AREA EMERGENCY

AS1.1 1 2 3 4 5 DEF

Any valid gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)

AS1.2 1 2 3 4 5 DEF

Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary

AS1.3 1 2 3 4 5 DEF

Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1)

OR

Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary

ALERT

AA1.1 1 2 3 4 5 DEF

Any valid gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.2 1 2 3 4 5 DEF

Any valid liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)

AA1.3 1 2 3 4 5 DEF

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)

UNUSUAL EVENT

AU1.1 1 2 3 4 5 DEF

Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.2 1 2 3 4 5 DEF

Any valid liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)

AU1.3 1 2 3 4 5 DEF

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)

2 — Onsite Rad Conditions & Spent Fuel Pool Events

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.80E+06 µCi/sec	2.24E+05 µCi/sec
GASEOUS	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	8.48E+04 µCi/sec
GASEOUS	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
GASEOUS	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	-----	-----	The lesser of *: 200 x calculated alarm values **OR** monitor upscale	The lesser of *: 2 x calculated alarm values **OR** monitor upscale
LIQUID	Service Water Effluent	-----	-----	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge **not** isolated

ALERT

AA2.1 1 2 3 4 5 DEF

Damage to irradiated fuel **OR** loss of water level (uncovering irradiated fuel outside the RPV) that causes **EITHER** of the following:

Valid RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr

OR

Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm

AA2.2 1 2 3 4 5 DEF

A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered

UNUSUAL EVENT

AU2.1 1 2 3 4 5 DEF

Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by **any** of the following:

- LI-86 (calibrated to 1001' elev.)
- Spent fuel pool low level alarm
- Visual observation

AND

Valid area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2

AU2.2 1 2 3 4 5 DEF

Unplanned valid area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels*

* Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value

3 — MCR/CAS Rad

ALERT

AA3.1 1 2 3 4 5 DEF

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Main Control Room (RM-RA-20)

OR

CAS

Notes

1. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

   If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values.
   (See EAL AS1.2/AG1.2.) Do **not** delay declaration awaiting dose assessment results.
2. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.
3. The Emergency Director should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.
4. Containment Closure is the action taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. Containment Closure requirements are specified in Administrative Procedure 0.50.5, Outage Shutdown Safety.
5. Manual scram methods for EAL SA2.1 and EAL SS2.1 are the following:
   - Reactor Scram push buttons
   - Reactor Mode switch in SHUTDOWN
   - Manual or auto actuation of ARI
6. See Table F-1, Fission Product Barrier Matrix, for possible escalation above the Unusual Event due to RCS Leakage.
7. If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.
8. The time required to establish the backfeed is likely longer than the specified time interval. If off-normal plant conditions have already established the backfeed, its power to the safety-related buses may be considered an offsite power source.