QUESTION 1

007EA1.03

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EA1.03	
	Importance Rating	4.2	

Proposed Question:

Plant conditions:

- The reactor has tripped due to a pressurizer steam space leak.
- Safety Injection has automatically initiated.

The **minimum** values allowed by EOP-E-0 to terminate Safety Injection are RCS pressure greater than __(1)__ PSIG and RCS subcooling based on core exit TCs greater than __(2)___ degrees F.

Which one of the following completes the statement above?

A	(1) (2)	1650 30
B	(1) (2)	1650 35
C	(1) (2)	1715 30
D	(1) (2)	1715 35

K/A Match Analysis

007 Reactor Trip, EA1.03

Ability to operate and monitor the following as they apply to a reactor trip: RCS pressure and temperature.

Requires the ability to monitor RCS pressure and temperature following a reactor trip and the operational implications of RCS pressure and temperature.

Answer Choice Analysis

A. (1) CORRECT (2) INCORRECT. Plausible because the Reactor Coolant Pumps are required to be secured if CETC are less than 30 degrees.

B. **(1) CORRECT (2) CORRECT**

C. (1) INCORRECT Plausible because 1715 PSIG is the Safety Injection set point.
(2) CORRECT

D. (1) INCORRECT (2) CORRECT

Technical Reference(s):

EOP-E-0 (REACTOR TRIP OR SAFETY INJECTION) Rev 2.

Proposed references to be provided to applicants during examination: None

Learning Objective:

EOP-E-0 Rev 2. Objective 6: Given a set of plant conditions, DETERMINE the actions directed by EOP-E-0, REACTOR TRIP OR SAFETY INJECTION

Question Source: New ____x____

Question History: New

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis _

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

…below

QUESTION 2

008AK2.03

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	008AK2.03
	Importance Rating	2.5

008AK2.03 – Knowledge of the interrelations between the PZR Vapor Space Accident and controllers and positioners.

Proposed Question:

Pressurizer Safety Valve RC-551A fails open with the following plant conditions:

- Mode 1
- Pressurizer level control is in a normal automatic alignment
- Charging pump speed controllers are in AUTO

Following the failure of RC-551A, pressurizer pressure stabilizes at 1100 psig.

Based on the conditions above, the output of LT-459 will cause the pressurizer level controller to demand:

a. raising the speed of the charging pumps.
b. de-energizing the backup heaters when level reaches 14.4%.
c. energizing the backup heaters when level reaches 27.2%.
d. no controller functions, since it is not the controlling channel.

Proposed Answer:

c. energizing the backup heaters when level reaches 27.2%.

Explanation:

Given the conditions presented in the question stem, the applicant needs to recognize that with a stuck open PZR safety valve AND pressurizer pressure stable at 1100 psig that the PZR will be at (or near) solid conditions. Using this knowledge, the applicant must then apply the conditions to the effects they would have on the PZR level control system.

a. **INCORRECT** – Since PZR level will be high given the plant conditions in the stem, the PZR level control system will send a demand signal to the charging pumps to lower charging pump speed. Plausible if the applicant misdiagnoses the given plant conditions and concludes that PZR level will be low.
b. **INCORRECT** – Since PZR level will be high given the plant conditions in the stem, the PZR level control system will send a demand signal to the backup heaters to energize. Plausible if the applicant misdiagnoses the given plant conditions and concludes that PZR level will be low.
c. **CORRECT** – Since PZR level will be high given the plant conditions in the stem, the PZR level control system will send a demand signal to energize all backup heaters when level reaches 27.2% (which is 5% higher than the no load T_{avg} value of 22.2%). The plant should be at no load conditions following an automatic reactor trip (most likely on low PZR pressure).
d. **INCORRECT** – PZR level is a selectable function, with the selected level transmitter sending input to the PZR level controller. The normally selected input comes from LT-459. Given the conditions in the stem, the applicant needs to determine that LT-459 is the controlling channel ("normal" configuration). Plausible since a different level transmitter can be selected to input to the PZR level controller.

Technical Reference(s): "Pressurizer/PRT System" SD-059, Revision 9 (see attached markups)

Proposed references to be provided to applicants during examination: none

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the conditions given in the stem indicate a PZR vapor space accident due to the failed open PZR safety valve. The applicant is then asked to determine the effects of the failed open safety valve on the PZR level controller (i.e the demand output from the controller to the backup heaters due to high level).

QUESTION 3

009EK2.03

K/A number and description:

009EK2.03. Knowledge of the interrelations between the small break LOCA and the following: S/Gs

Question:

Given the following plant conditions:

- The unit was initially at 100% power.
- A LOCA has occurred.
- RCS pressure is 1300 psig and lowering.
- Tave is 548 F.
- Containment pressure peaked at 4.2 psig.
- The crew is performing actions of EPP-8, Post LOCA Cooldown and Depressurization.

As directed by EPP-8 which one of the following identifies the method that will be used to initiate cooldown of the RCS?

Initiate cooldown using the...

A. steam dumps at the maximum achievable rate.

B. steam dumps at no greater than 100°F/hr in the last 60 minutes.

C. Steam Line PORVs at the maximum achievable rate.

D. Steam Line PORVs at no greater than 100°/hr in the last 60 minutes.

Distractor Analysis:

A. Incorrect - Steam dumps are available but max cooldown achievable would be greater than 100 F in one hour which is incorrect. Plausible since several EOPs procedures direct the operator to cooldown at maximum achievable rate.

B. CORECT - Steam dumps are available due to the MSIVs remaining open and directed to be used per EPP-8. The cooldown rate is correct.

C. Incorrect. The first part of distractor is incorrect. MSIVs remain open and steam dumps will be used vs. SG PORVs per EPP-8. The second part is incorrect; the max cooldown achievable would be greater than 100 F in one hour which is incorrect. Plausible since several EOPs procedures direct the operator to cooldown at maximum achievable rate.

D. Incorrect. . The first part of distractor is incorrect. MSIVs remain open and steam dumps will be used vs. SG PORVs per EPP-8. The second part is correct.

Reference(s)

- EPP-8, Post LOCA Cooldown and Depressurization, Rev. 21

K/A Match discussion:

The question matches the K/A because plant conditions are related to a small break LOCA and the applicant determines how these conditions impact cooldown options via the S/Gs.

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

Question Source:	Bank:	_______	
	Modified Bank:	___X___	(Note changes or attach parent)
	New:	_______	

Comment: Bank source was RNS NRC exam 2011-302, Q 3. Modified plant conditions be more aligned with a small break vs. a large break LOCA. Changes resulted in a different correct answer because a phase B no longer existed and no Main Steam Isolation occurred.

SRO Only Basis: N/A

QUESTION 4

011EG2.4.21

Large Break LOCA /3
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
RO 4.0/SRO 4.6
CFR 41.7/43.5

Unit 2 has suffered a Large Break LOCA.
Restoration efforts are in progress with the following plant conditions:

- Core Exit Thermocouples are currently reading 720^{o}F
- Full Range RVLIS is 45%
- CV Pressure is currently 11 psig
- CV Sump Level is 380 inches and rising
- CV Radiation Levels are > 5 R/hr
- "A" and "B" CV Spray Pumps have tripped

Given:

- FRP-C.1, Response to Inadequate Core Cooling
- FRP-C.2, Response to Degraded Core Cooling
- FRP-J.1, Response to High Containment Pressure
- FRP-J.2, Response to Containment Flooding

Which ONE of the following completes the statement below?

Based on the indications above, which Function Restoration Procedures (FRP) have met entry criteria in accordance with Critical Safety Function Status Trees (CSFT):

A. FRP-C.1 and FRP-J.1

B. FRP-C.1 and FRP-J.2

C. FRP-C.2 and FRP-J.1

D. FRP-C.2 and FRP-J.2

Distractor Analysis

A. Incorrect. Plausible if applicant misinterprets supplied RVLIS Full Range and CV pressure indications. (Ref. CSFST, Pages 4 and 8)

B. Incorrect. Plausible if applicant misinterprets supplied RVLIS Full Range indication. (Ref. CSFST, Page 4).

C. Incorrect. Plausible if misinterprets supplied CV pressure indication. (Ref. CSFST, Page 8)

D. Correct. Given information provides applicant success path

References

- CSFST, Critical Safety Function Status Trees, Revision 6, Pages 4 and 8

KA Match

Applicant is supplied conditions indicative of a recent Large Break LOCA and given these parameters must predict the correct critical safety function path and priority of entry.

Cognitive Level

High, due to evaluations of multiple parameters to arrive at correct conclusion.

Source of Question

Modified, 2013-301 NRC Exam

SRO Only Basis

Not applicable

QUESTION 5

022AA1.09

APE022 AA1.09
Loss of Reactor Coolant Makeup
Ability to operate and/or monitor the following as they apply to Loss of Reactor Coolant Makeup:
RCP seal flows, temperatures, pressures, and vibrations. 3.2/3.3 (41.7, 45.5, 45.6)

- The plant is at 100% power
- At 0200 the Reactor Operator determined that Seal Injection flow was lost to 'A' RCP
- The following 'A' RCP conditions existed at that time:

Pump Bearing Temperature	Shaft Vibration	#1 Seal Leakoff Temperature
221ºF	12.4 mils	223ºF

- It is now 0300 and the following 'A' RCP conditions exist:

Pump Bearing Temperature	Shaft Vibration	#1 Seal Leakoff Temperature
226ºF	14.4 mils	229ºF

With the above stated conditions, which statement describes the required actions for 'A' RCP, if any?

A. RCP operation may continue. There are no parameters requiring immediate RCP shutdown.

B. RCP must be shut down due to pump bearing temperature exceeding operating limits.

C. RCP must be shut down due to pump shaft vibration exceeding operating limits.

D. RCP must be shut down due to #1 seal leakoff temperature exceeding operating limits.

Distractor Analysis

A. Incorrect. ‘A’ RCP pump bearing temperature exceeds the 225ºF limit and must be shut down.

B. **CORRECT**. RCP Pump Bearing temperature limit is 225ºF per OP-101 Table 1, RCP Operating Limits. AOP-018 has you trip the reactor and stop the affected RCP(s).

C. Incorrect. Shaft vibration criteria for pump shutdown is 20 mils, or >15 mils and rising at 1 mil per hour. Vibration has risen 2 mils in 1 hour, but are still below the value of 15 mils where the rate would apply.

D. Incorrect. Seal leakoff temperature limit is 235ºF. Plausible if the applicant juxtaposes the values for pump bearing (225F) and seal leakoff (235F). This almost makes more sense given the system design of the pump bearing being upstream of the seal leakoff temperature measurement. Applicant could rationalize that you’d want seal leakoff to alarm 10º earlier than the pump bearing, rather than 10º later.

References:

- AOP-018, REACTOR COOLANT PUMP ABNORMAL CONDITIONS, Rev. 28
- AOP-018 Basis Document, Rev. 28
- OP-101, REACTOR COOLANT SYSTEM AND REACTOR COOLANT PUMP STARTUP AND OPERATION, Rev. 69

KA Match:

The KA is matched because the applicant must demonstrate the ability to monitor/interpret RCP seal flows, temperatures, pressures, and vibrations as they apply to Loss of Reactor Coolant Makeup, and then choose whether to continue to operate RCP(s) or not.

Cognitive Level: **Low**

Source of Question: Indian Point 2003 NRC exam. Removed frame vibration and added Pump Bearing temperature. Changed correct answer from Seal Outlet temperature to Pump Bearing temperature. Changed all values to be RNP-specific.

QUESTION 6

025AA2.07

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	AA2.07	
	Importance Rating	3.4	

Proposed Question:

Unit 1 Plant Conditions:

- Shutdown with the RHR System aligned for core cooling.
- RCS level is at (-) 69 inches and stable as read on LI-403 and LI-404.
- Steam Generator manway removal in progress.
- The "A" RHR pump is running.
- RHR System total flow is oscillating from 2800 to 3500 gpm.
- "A" RHR pump amps are oscillating.
- RHR HX LO FLOW annunciator, APP-001-A7 alarming.

Based on the above conditions, which of the following identifies the **first** correct action to mitigate the event as directed by AOP-020 (LOSS OF RESIDUAL HEAT REMOVAL)?

A	Raise RCS level.
B	Reduce RHR flow.
C	Stop the "A" RHR pump.
D	Start the “B” RHR pump.

K/A Match Analysis

025 Loss of Residual Heat Removal System (RHRS), AA2.07

Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation.

Requires the ability to interpret plant indications and determine a Loss of Residual Heat Removal due to pump cavitation.

Answer Choice Analysis

E. Incorrect. Plausible due to step 15 and its RNO of AOP-020 require isolating letdown and increasing charging flow.

F. **Correct. Step 23.a of AOP-020 Section E requires reduction of RHR flow to 1500gpm in an attempt to eliminate "A" RHR pump cavitation. This is the first correct action from AOP-020 when routing correctly through the procedure.**

G. Incorrect. Plausible due to step 2 and step 22.a RNO of AOP-020 Section E require stopping the running RHR pump if RCS level is not above (-) 72 inches.

H. Incorrect. Plausible due to step 7 of AOP-020 requires a check of RHR pump status and step 7.e. RNO requires starting the standby RHR pump.

Technical Reference(s):

AOP-020 (LOSS OF DECAY HEAT REMOVAL CAPABILITY) Rev 40

GP-008 (DRAINING THE REACTOR COOLANT SYSTEM), Revision 81

Proposed references to be provided to applicants during examination: None

Learning Objective:

Lesson plan AOP-020 Procedure Review Rev 38; objective 3. Given a set of plant conditions, DETERMINE the appropriate plant response and operator actions in accordance with AOP-020.

Question Source: Modified Bank # Surry 2012-301 Q.5 (parent attached)

Question History: Last NRC Exam: Modified from Surry 2012-301 Q.5

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 7

026AK3.03

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	026AK3.03
	Importance Rating	4.0

026AK3.03 – Knowledge of the reasons for the guidance actions contained in EOPs for Loss of CCW as they apply to the Loss of Component Cooling Water.

Proposed Question:

The control room operating crew has entered FRP-C.2 “Response to Degraded Core Cooling” and the following plant conditions exist:

- ‘A’ RCP is running
- ‘B’ and ‘C’ RCPs have been secured
- Component cooling water flow to the ‘A’ RCP motor has just been lost
- ‘A’ RCP upper thrust bearing temperature increased to 205 °F

Based on the above conditions, which one of the following choices:

(1) describes whether the ‘A’ RCP is required to be immediately secured

(2) the reason for either securing the ‘A’ RCP or allowing it to run?

	(1)	(2)
a.	‘A’ RCP is required to be immediately secured	bearing and motor damage could occur
b.	‘A’ RCP is required to be immediately secured	RCP ‘A’ could be pumping two-phase flow through the core
c.	‘A’ RCP is **NOT** required to be immediately secured	PZR spray is being supplied by the ‘A’ loop
d.	‘A’ RCP is **NOT** required to be immediately secured	RCP ‘A’ is providing single or two-phase flow through the core

Proposed Answer:

d.	‘A’ RCP is **NOT** required to be immediately secured	RCP ‘A’ is providing single or two-phase flow through the core

Explanation:

Given that the plant operating crew has entered FRP-C.2 with a concurrent loss of CCW to the only running RCP, the applicants need to recall that, while in FRP-C.2, RCPs should not be tripped until specifically directed in the procedure regardless of the status of support conditions to the respective RCP.

a. **INCORRECT** - Even though a loss of CCW to the ‘A’ RCP may in fact lead to damage of the motor, per the basis for FRP-C.2, RCPs should not be tripped until the procedure directs securing the pumps as doing so may result in an inadequate core cooling condition. RCP trip criteria DO NOT apply.
b. **INCORRECT** - Per the basis for FRP-C.2, RCPs should not be tripped until the procedure directs securing the pumps as doing so may result in an inadequate core cooling condition. RCP trip criteria DO NOT apply. Even if the ‘A’ RCP is pumping two-phase coolant through the core, it should still provide some cooling. Plausible in some small break LOCA scenarios.
c. **INCORRECT** - FRP-C.2 does offer guidance to NOT trip the ‘A’ RCP, but it is not to allow better pressure control of the Reactor Coolant System. The PZR spray lines are on the ‘B’ and ‘C’ loops, therefore the ‘A’ RCP will not aid in maintaining spray flow for pressure control.
d. **CORRECT** – Per the basis for FRP-C.2, RCPs should not be tripped until the procedure directs securing the pumps as doing so may result in an inadequate core cooling condition. RCP trip criteria DO NOT apply.

Technical Reference(s): FRP-C.2, Revision 16, “Response to Degraded Core Cooling”
FRP-C.2, Revision 16, “FRP-C.2 Basis Document”
SD-059, Revision 9, “Pressurizer/PRT System”
(see attached markups)
AOP-014, Revision 36, “Component Cooling Water System Malfunction”

Proposed references to be provided to applicants during examination: none

Question Source: North Anna Bank #2540 (see attached bank question)

Question History: Unknown

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

K/A match: The K/A is matched since it requires the applicant to demonstrate understanding of the reasons for not tripping RCPs during a loss of CCW while in FRP-C.2 “Response to Degraded Core Cooling.”

QUESTION 8

027AG2.4.49

K/A number and description:
027AG2.4.49. Ability to perform without reference to procedures those actions that require immediate operation of system components and controls: Pressurizer Pressure Control System Malfunction.

Question:
Given the following:

- The plant is operating at 100% RTP.
- A failure of the pressurizer pressure Master Controller, PC-444J, results in an actual pressure decrease to 2185 psig and lowering.

If PC-444J is placed in manual per the IMMEDIATE ACTIONS of AOP-019, Malfunction of RCS Pressure Control, the controller's ___(1)_____ arrow would be initially depressed to regain pressure control.

If spray valve PCV-445A would not close and the reactor was tripped due to lowering pressure, then the minimum RCP(s) required to be secured per EOP-E-0, Reactor Trip or Safety Injection, to stop spray flow is/are RCP ___(2)____.

	(1)	(2)
A.	down	B and C
B.	down	C
C.	up	B and C
D.	up	C

Proposed correct answer: A

Distractor Analysis:

A. Correct. The first part is correct. After de-selecting auto to place in manual, the master controller's down arrow will be required to be pressed to increase heater output and decrease spray demand to raise actual pressure. The second part is correct as securing both RCP B and C is directed to decrease spray flow through PCV-455A.

B. Incorrect. The first part is ~~in~~correct. The first part is correct. After de-selecting auto to place in manual, the master controller's down arrow will be required to be pressed to increase heater output and decrease spray demand to raise actual pressure. The second part is incorrect as securing both RCP B and C is directed to decrease spray flow through PCV-455A. The answer is plausible because RCP C only would be the correct answer is spray valve PCV-455B were stuck open.

C. Incorrect. The first part is incorrect. After de-selecting auto to place in manual, pressing the master controller's up arrow would decrease heater output and increase spray demand thereby causing the actual pressure to continuing lowering. This answer is plausible because the operator has to understand the controller action and whether the arrows are referenced to raising or lowering pressure or the heater/spray action. The second part is correct as securing both RCP B and C is directed to decrease spray flow through PCV-455A.

D. Incorrect. The first part is incorrect. . After de-selecting auto to place in manual, pressing the master controller's up arrow would decrease heater output and increase spray demand thereby causing the actual pressure to continuing lowering. This answer is plausible because the operator has to understand the controller action and whether the arrows are referenced to raising or lowering pressure or the heater/spray action. The second part is incorrect as securing both RCP B and C is directed to decrease spray flow through PCV-455A. The answer is plausible because RCP C only would be the correct answer is spray valve PCV-455B were stuck open.

Reference(s)

- AOP-019, MALFUNCTION OF RCS PRESSURE CONTROL, R18
- AOP-019-BD, AOP-019 Basis Document
- EOP-E-0, REACTOR TRIP OR SAFETY INJECTION
- SD-059, Pressurizer/PRT System Description, R9

K/A Match discussion:

Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	x	

SRO Only Basis: N/A (or) See attached justification

QUESTION 9

029EG2.4.31

029EG2.4.31
ATWS /1
Knowledge of annunciator alarms, indications or response procedures
RO 4.2/SRO 4.1
CFR 41.10/45.3

A primary system pipe break with a failure of the reactor to trip (ATWS) has resulted in the following plant conditions:

- Reactor Power is 12% and decreasing
- CV Pressure is 12 psig and increasing

For the above conditions, FRP-S.1 “Response to Nuclear Power Generation/ATWS” states that RCP’s ____(1)____ be tripped.

Per FRP-S.1, if the Main Turbine is unable to be tripped, uncontrolled RCS cooldown is prevented by **first** performing ____(2)____ .

	(1)	(2)
A.	should	a Turbine runback
B.	should	a closure of all MSIV’s
C.	should NOT	a Turbine runback
D.	should NOT	a closure of all MSIV’s

Distractor Analysis

A. Incorrect. Answer is plausible due to being partially correct. Normally with a phase B signal in, RCP's would be required to be secured; however, due to the given conditions, plant procedures direct RCP's to remain in operation. RCP operation during ATWS conditions when > 5% reactor power is beneficial even if all normal running conditions are not satisfied. (Ref FRP-S.1-BD, Page 46 and FRP-S.1, Page 3)

B. Incorrect. Plausible if applicant misapplies given information to answers, see "A" and "D" distractor analyses.

C. Correct. Given information provides applicant success path.

D. Incorrect. Answer is plausible due to being partially correct. FRP-S.1 specifies that in the event of a failure of the Main Turbine to trip, operators are to first attempt manual runback of the Main Turbine. (Ref FRP-S.1, Page 4 and FRP-S.1-BD, Page 48)

References

- FRP-S.1, Revision 20, Pages 3 and 4
- FRP-S.1-BD, Revision 20, Pages 46 and 48

KA Match

Applicant is supplied conditions indicative of a primary system break concurrent with an ATWS with a Main Turbine failure to trip and must predict crew response.

Cognitive Level

Low

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 10

038EA1.10

EPE038 EA1.10
Steam Generator Tube Rupture
Ability to operate and monitor the following as they apply to a SGTR:
Control room radiation monitoring indicators and alarms. 3.7*/3.7 (41.7, 45.5, 45.6)

Given the following plant conditions:

- The plant is at 100% RTP.
- The following Radiation Monitors go into alarm:
  - Condenser Air Ejector Gas Monitor, R-15
  - Condensate Polisher Waste Effluent Monitor, R-37
  - S/G Sample Radiation Monitor, R-19A
  - "A" Main Steam Line N-16 Detector, R-24A

Which ONE (1) of the following describes the correct response to the above given conditions?

A. Condensate Polishing Building Sump pumps TRIP.
V1-31, Blowdown Isolation Valve to Catch Basin, CLOSES.

B. FCV-1933A and B, S/G A Blowdown Sample Isolation Valves, CLOSE.
Condensate Polishing Building Sump pumps TRIP.

C. FCV-1933A and B, S/G A Blowdown Sample Isolation Valves, CLOSE.
RCV-10549, Condensate Polisher Discharge to Storm Drains, CLOSES.

D. V1-31, Blowdown Isolation Valve to Catch Basin, CLOSES.
RCV-10549, Condensate Polisher Discharge to Storm Drains, CLOSES.

Distractor Analysis

A. Incorrect. 1st part is incorrect: the sump discharge valve closes, but the sump pumps do not trip. Plausible because in this situation you're dead-heading a pump, so an applicant might think the signal also trips the pumps. 2nd part is incorrect but plausible because V1-31 does close if all three R-19s are in alarm.

B. Incorrect. 1^{st} part is correct. 2^{nd} part incorrect as discussed above.

C. **CORRECT**. An R-19 alarming will close its Steam Generator Blowdown and Sample valves, as well as its blowdown flow control valve, FCV-4204x. R-37 alarming will close the Condensate Polisher Building sump discharge valve, but not trip any running sump pumps (there are two).

D.. Incorrect. 1^{st} part incorrect as discussed above, 2^{nd} part correct.

References:
ST-020, Steam Generator Blowdown System, Rev. 3
ST-050, Condensate Polishing System, Rev. 2

KA Match:
The KA is matched because an applicant must recall automatic actions caused by alarms on two different radiation monitors related to a SGTR.

Cognitive Level: **Low**

Source of Question: RNP exam bank question 5749.

QUESTION 11

040AK3.04

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	K3.04	
	Importance Rating	4.5	

Proposed Question:

All Steam Generators are faulted.

___________is the basis for a minimum 80 gpm indicated AFW flow to each Steam Generator per EPP-16 (Uncontrolled Depressurization of All Steam Generators).

Which one of the following completes the above sentence?

A	Preventing Steam Generator dryout
B	Prevent water hammer in the feed rings
C	Meeting minimum heat sink flow requirements
D	Ensure the feed lines stay warm to prevent excessive thermal shock to the feed lines during recovery actions

Proposed Answer: **A**

K/A Match Analysis

Requires knowledge of the reason for actions contained in the EOP for a steam line rupture.

Answer Choice Analysis

A. CORRECT. 80 gpm is the minimum verifiable flow rate to a steam generator. This ensures a nominal flow rate of 25 gpm, including detector uncertainties, to prevent dryout and thermal shock to the S/G.

B. INCORRECT. Water hammer is plausible because design changes such as J-tube installation on feed-rings have occurred over the years to mitigate water hammer concerns and water hammer continues to be a concern.

C. INCORRECT. Plausible because in other EOP circumstances minimum AFW flow rates to meet minimum heat sink requirements are identified.

D. INCORRECT. Plausible because in other EOP circumstances, thermal shock to the feed lines is a concern.

Technical Reference(s):

EPP-16 UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, rev 21
EPP-16 BD (Basis Document) rev 21
Lesson EPP-16 rev 21

Proposed references to be provided to applicants during examination: None

Learning Objective:
Objective 9: STATE/IDENTIFY the Basis for the overall mitigating strategies, steps, notes, cautions, and attachments for EPP-16, Uncontrolled Depressurization of All Steam Generators

Question Source:

Modified Bank: Surry 2004-301, Q.74 (modified to fit Robinson and modified distractors, parent attached).

Question History: Last NRC Exam: Surry 2004-301, Q.74 which came from modified Surry ILT Bank Question #1010.

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis _

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

Attached

QUESTION 12

054AA2.03

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	054AA2.03
	Importance Rating	4.1

054AA2.03 – Ability to determine and interpret the conditions and reasons for AFW pump startup as they apply to the Loss of Main Feedwater (MFW)

Proposed Question:

The unit has entered E-0, “Reactor Trip or Safety Injection,” from AOP-010, “Main Feedwater / Condensate Malfunction,” due to a trip of both main feedwater pump breakers.

- Steam generator narrow range levels following the reactor trip <u>lowered to and are currently</u>:
  - ‘A’ S/G = 16% and stable
  - ‘B’ S/G = 17% and stable
  - ‘C’ S/G = 17% and stable
- Operators are currently performing E-0, Step 6, “Check AFW Pumps – RUNNING.”

With respect to the Auxiliary Feedwater (AFW) system and the above plant conditions:

(1) What is the status of the SDAFW pump?
(2) What actions are required in accordance with E-0?

	(1)	(2)
a.	OFF	manually start the SDAFW pump
b.	OFF	check AFW header section valves FULL OPEN
c.	ON	manually stop the SDAFW pump
d.	ON	check SDAFW pump discharge valves FULL OPEN

Proposed Answer:

b. OFF check AFW header section valves FULL OPEN

Explanation:

This question requires the applicant to recall the automatic start signals for both the MDAFW and SDAFW pumps and apply the signals to conditions following a manual reactor trip as directed in AOP-010. The applicant also has to apply knowledge/reasons for required actions from E-0 based on the conditions stated in the question stem.

a. **INCORRECT** – The first part of the answer is correct; due to the main feedwater pump breakers tripping, the MDAFW pumps are designed to auto start. There have been no auto start signals for the SDAFW pump (low-low level on 2/3 SGs, undervoltage on 4160VAC buses 1 and 4) AND per EOP-E-0, Step 6b, the SDAFW pump is only directed to be started if 2 S/G's have levels less than 16%. Plausible if the applicant does not remember the procedural guidance in E-0 for manually starting the SDAFW pump.
b. **CORRECT** - The first part of the answer is correct; due to the main feedwater pump breakers tripping, the MDAFW pumps are designed to auto start. There have been no auto start signals for the SDAFW pump (low-low level on 2/3 SGs, undervoltage on 4160VAC buses 1 and 4). The second part of the answer is also correct; since manual start of the SDAFW pump is not required, the operators are then directed to Step 7 of E-0 which directs verifying proper valve alignment of AFW (which includes the AFW header section valves being FULL OPEN).
c. **INCORRECT** - The first part of the answer is incorrect; there has been no auto start signal for the SDAFW pump. The second part of the answer would be correct if the SDAFW pump was running. Plausible if the applicant determines that the SDAFW pump is running based on the conditions stated in the stem of the question.
d. **INCORRECT** – The first part of the answer is incorrect; there has been no auto start signal for the SDAFW pump. The second part of the answer would be correct if the SDAFW pump was running. Plausible if the applicant determines that the SDAFW pump is running based on the conditions stated in the stem of the question.

Technical Reference(s): AOP-010 "Main Feedwater / Condensate Malfunction", Revision 30
E-0 "Reactor Trip or Safety Injection", Revision 2
SD-042 "Auxiliary Feedwater System", Revision 13
(see attached markups)

Proposed references to be provided to applicants during examination: none

Question Source: New

Question History: N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant is required to demonstrate knowledge of the conditions requiring automatic startup of the MDAFW and SDAFW pumps. The second part of the question requires the applicant to know the reasons for taking action with respect to operation of the AFW pumps / system.

QUESTION 13

055EK1.02

K/A number and description:

055EK1.02: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling.

Question:

A cooldown of < 100 °F/hr is in progress per EPP-1, Loss of All AC, using Steam Line PORVs.

EPP-1 states to both:

- depressurize intact S/Gs at maximum rate to __(1)__ psig, **and**
- during the depressurization check RCS cold leg temperature greater than __(2)___ °F.

	(1)	(2)
A.	140	320
B.	140	290
C.	240	320
D.	240	290

Distractor Analysis:

A. Incorrect. The first part of the answer is incorrect; however, 140 psig is plausible because it is the nominal SG pressure to preclude injection of accumulator nitrogen into the RCS before the 100 psi margin is added for controllability. The second part of the answer is correct and is included in the step per the basis which states that RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function.

B. Incorrect. The first part of the answer is incorrect however 140 psig is plausible because it is the nominal SG pressure to preclude injection of accumulator nitrogen into the RCS before the 100 psi margin is added for controllability. The second part of the answer is incorrect; however, it is plausible because it is the RCS cold leg temperature setpoint in the CSF-4, RCS Integrity Status Tree which would complete Orange Path criteria and require transition to FRP-P.1

C. Correct. The first part of the answer is correct because it is less than 240 psig which is the procedural value given to preclude injection of accumulator nitrogen with a 100 psi margin added for controllability. The second part of the answer is correct and is included in the step per the basis which states that RCS cold leg temperatures should be monitored during SG depressurization to ensure that the depressurization does not impose a challenge to the Integrity Critical Safety Function.

D. Incorrect. The first part of the answer is correct because it is less than 240 psig which is the procedural value given to preclude injection of accumulator nitrogen with a 100 psi margin added for controllability. The second part of the answer is incorrect; however, it is plausible because it is the RCS cold leg temperature setpoint in the CSF-4, RCS Integrity Status Tree which would complete Orange Path criteria and require transition to FRP-P.1

Reference(s)

- EPP-1, Loss of All AC Power (aka ECA 0.0)
- EPP-1-BD, EPP-1 Basis Document
- CSF-4, RCS Integrity

K/A Match discussion:

The question requires knowledge of limits that are imposed in EPP-1 during a natural circulation cooldown due to a station blackout.

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

Question Source: Bank: ______
Modified Bank: _______ (Note changes or attach parent)
New: ___X___

SRO Only Basis: N/A

QUESTION 14

056AK1.04

Loss of Off-site Power /6
Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of saturation conditions, implication for the systems
RO 3.1/SRO 3.2
CFR 41.8/41.10

Initial Conditions:

- Unit 2 was at 100% power
- A Loss of all Offsite Power (LOOP) has occurred
- Operators entered EPP-5, "Natural Circulation Cooldown"

Current Conditions:

- Emergency Diesel Generators continue to operate normally
- Operators have just completed the step in EPP-5 to check cool down rate required – GREATER THAN limits, and have determined that a transition to EPP-6, "Natural Circulation Cool down with Steam Void in Vessel" is not required
- CRDM Cooling Fan HVH-5A is running
- CRDM Cooling Fan HVH-5B is NOT running

Based on the current conditions, EPP-5 states that both: (1) maintaining RCS sub-cooling __________ 100 ^{o}F and (2) cool down less than __________ ^{o}F/hr precludes formation of a steam void in the vessel upper head.

	(1)	(2)
A.	at	10
B.	at	25
C.	greater than	10
D.	greater than	25

<u>Distractor Analysis</u>

A. <u>Correct</u>. Given information provides applicant success path.

B. <u>Incorrect</u>. Answer is plausible due to being partially correct. The 25 ^{o}F/hr cool down rate is applicable when cooling down to Cold Shutdown to maintain conformance with T.S. limits. (Ref EPP-5, Page 8)

C. <u>Incorrect</u>. Answer is plausible due to being partially correct. Maintaining RCS sub-cooling specifically at 100 ^{o}F with the correct cool down rate assures compliance with Technical Specifications. (Ref EPP-5-BD, Page 23)

D. <u>Incorrect</u>. Plausible if applicant misapplies incorrect answers to question, see above.

<u>References</u>

- EPP-5, Revision 18, Pages 8 & 12
- EPP-5-BD, Revision 18, Page 23

<u>KA Match</u>

Applicant is supplied conditions where limits must be produced that ensure saturation conditions are prevented from forming during a Loss of all Offsite Power.

<u>Cognitive Level</u>

High, due to evaluation of multiple sub-cooling and cool down rate limits to arrive at correct conclusion.

<u>Source of Question</u>

New

<u>SRO Only Basis</u>

Not applicable

QUESTION 15

057AK3.01

APE057 AK3.01
Loss of Vital AC Electrical Instrument Bus
Knowledge of the reasons for the following responses as they apply to Loss of Vital AC Instrument Bus:
Actions contained in EOP for loss of vital ac electrical instrument bus.
4.1/4.4 (41.5, 41.10, 45.6, 45.13)

Given the following:

- Unit was at 100% power.
- Loss of a Vital Instrument Bus occurred
- The control room crew has just completed the Immediate Actions of AOP-024, LOSS OF INSTRUMENT BUS.

The Immediate Action, “Place the Main Turbine in Manual,” is procedurally directed to be performed for loss of ANY Instrument Bus, without taking time at that point to diagnose which bus was lost.

The BASIS for the action is to mitigate the effects of a loss of Instrument Bus __**(1)**__.

This is because a continuous turbine Load Reference runback would occur if the turbine were in Automatic, making an automatic reactor trip unavoidable because __**(2)**__**.**

	(1)	(2)
A.	1	all Pressurizer spray valve control is lost
B.	1	all Feedwater Regulating Valve M/A stations are lost
C.	4	all Pressurizer spray valve control is lost
D.	4	all Feedwater Regulating Valve M/A stations are lost

Distractor Analysis

A. Incorrect. Loss of IB1 does cause all Pressurizer spray valve control to be lost (PC-444J, PZR Pressure Controller, and the individual spray valve controllers PC-444G & PC-444H "lock up"), which leads to the second part also being a true statement. But it's IB4, not IB1, that causes a continuous Load Reference runback and necessitates taking turbine controls to Manual.

B. Incorrect. The first part is plausible with the second part because on loss of IB1, FRV 'A' Automatic control is lost, so an applicant might confuse that with loss of all FRVs.

C. Incorrect. First part is correct. According to the Basis Document for AOP-024, loss of IB4 will cause a continuous Load Reference runback, and if the turbine were in Auto it wouldn't be survivable with power also lost to all three FRV controllers. The second part is plausible because loss of IB4 causes loss of "all automatic PZR Pressure Controllers", but not *manual* control.

D. **CORRECT**. First part correct as discussed above. Second part is correct per the Discussion section of the Basis Document for AOP-024: "IB 4 also provides M/A power to all three FW Reg Valve controllers on the RTGB (FC-478, 488, and 498). On a loss of IB 4, these controllers will fail as-is and lockup the FW Reg valves at their current position." Also from the Step Description for Step 1: "This is only needed for a loss of Instrument Bus 4..., however since the steps for diagnosing which instrument bus is lost have not yet been completed, the Turbine is placed in manual for all buses..."

References:
AOP-024, Loss of Instrument Bus, Revision 38
AOP-024, Loss of Instrument Bus, Basis Document, Revision 38

KA Match:
The KA is matched because the applicant is asked the reasons for a specific step in an AOP, and has to recall not only which instrument bus the action is specifically for, but also why. The KA is a soft match due to AOP actions vs the EOP actions required by the KA. At Robinson, Vital Power Supply malfunctions are typically addressed in AOPs.

Cognitive Level: **High**

Source of Question: **New**

QUESTION 16

065AA2.06

Examination Outline Cross-Reference:	Level	**RO**	SRO
	Tier #	1	
	Group #	1	
	K/A #	AA2.06	
	Importance Rating	3.6	

Proposed Question:

Given the following plant conditions:

- Plant is at 35% RTP.
- APP-002-E7 (INST AIR COMP D TRIP) in alarm.
- APP-002-F7 (INST AIR HDR LO PRESS) in alarm.
- AOP-017 (LOSS OF INSTRUMENT AIR) is in progress.
    - Station Air and Instrument Air have been cross-connected.
    - Transition has been made to AOP-017; Section A (Modes 1 AND 2).
    - Instrument Air pressure is 63 psig.
    - "B" and "C" S/Gs Levels are at 41% and slowly lowering.

The operating crew is required to_____ while continuing in AOP-017.

Which one of the following completes the statement above?

A	dispatch operator(s) to cross-connect Station Air and Construction Air
B	trip the reactor and Go to EOP-E-0 (Reactor Trip Or Safety Injection)
C	trip the turbine and implement AOP-007 (Turbine Trip below P-8)
D	lower turbine load as necessary to maintain feed and steam flows matched

Proposed Answer: **B**

K/A Match Analysis

065 Loss of Instrument Air, AA2.06

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

Requires knowledge of the reactor trip requirements for decreasing IA pressure contained in AOP-017, Loss of Instrument Air.

Answer Choice Analysis

A. INCORRECT. AOP-017 does not direct cross-connecting station air and construction air. Plausible because AOP-017 does direct cross connecting IA and SA. Construction Air can be cross connected with SA, and would supply more are to IA if done, however, it is typically done during an outage or when there is a large demand on station air.

B. CORRECT. Step 2 of AOP-017, Attachment A, asks the operator to check any S/G level control "Adversely affected by loss of IA." This would be answered "YES." Step 3 has the operator trip the reactor and go to EOP-E-0 and concurrently continue in AOP-017.

   NOTE: The plausibility of the distractors is increased by the fact that AOP-017 Step 2 is a continuous action step and it directs a reactor trip if IA pressure decreases to 60 psig. Applicant may not choose the correct answer because IA header pressure is at 63 psig.

C. INCORRECT. Plausible because AOP-017 directs that the reactor be tripped if S/G level is adversely affected by loss of IA if in Mode 1 or 2. Candidate may think that only a turbine trip is required since power is less than P-8. A Turbine Trip would not result in a Reactor Trip below P-8 (40%).

D. INCORRECT. Plausible because lowering turbine load will assist in matching steam flow and feed flow to raise SG level but this is not a mitigation strategy specified in AOP-017.

Technical References:
AOP-017 (Loss of Instrument Air) rev 40.
OP-905 (Instrument and Station Air System) rev 110.

Proposed references to be provided to applicants during examination: None

Learning Objective:
Lesson AOP-017 rev 40, Objective 2:
STATE/IDENTIFY the following for AOP-017:

a. Actions taken at key Instrument Air pressures
b. Fail positions for major air operated valves
c. Conditions requiring a reactor trip

Question Source: Robinson 2011-302 exam Q16 (updated to match current procedures). Parent question attached.

Question History: Last NRC Exam: 2011-302, Q16. Prior to that: 2007-301.

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

10 CFR Part 55 Content:	55.41	X
	55.43	_

Supporting References

Attached

QUESTION 17

WE04EK2.2

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	WE04EK2.2
	Importance Rating	3.8

WE04EK2.2 – Knowledge of the interrelations between the LOCA outside containment and the facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question:

In accordance with EPP-20, "LOCA Outside Containment":

(1) The **FIRST** flowpath to be isolated from the RCS when attempting to isolate an RCS leak outside containment will be associated with _____(1)_____.

AND

(2) The RCS parameter to be monitored to confirm that the RCS leak has or has NOT been isolated is _____(2)_____.

	(1)	(2)
a.	RHR	RCS pressure
b.	RHR	PZR level
c.	RCP Seal Return	RCS pressure
d.	RCP Seal Return	PZR level

Proposed Answer:

a. RHR RCS pressure

Explanation:

This question requires the applicant to demonstrate knowledge of the major actions and diagnosis guidance contained in EPP-20 “LOCA Outside Containment.”

a. **CORRECT** – Steps 1 through 4 of EPP-20 direct the operators to isolate connections between RHR and the RCS. As different leak locations are isolated throughout EPP-20, the operators determine whether or not the leak is isolated by checking if RCS pressure is rising, which is typically indicative of leak isolation.
b. **INCORRECT** – Steps 1 through 4 of EPP-20 direct the operators to isolate connections between RHR and the RCS. As different leak locations are isolated throughout EPP-20, the operators determine whether or not the leak is isolated by checking if RCS pressure is rising, which is typically indicative of leak isolation. The second part of the distractor is plausible since, per the basis document of EPP-20, certain leak sizes and locations may require more time for RCS re-pressurization to take place after leak isolation, therefore other means of verifying break isolation should be checked (i.e. PZR level). However, the procedure steps in EPP-20 do not direct the operators to use any indications other than RCS pressure.
c. **INCORRECT** – Steps 1 through 4 of EPP-20 direct the operators to isolate connections between RHR and the RCS. Plausible, since EPP-20 does direct the operators to isolate RCP seal return flow in Step 6. As different leak locations are isolated throughout EPP-20, the operators determine whether or not the leak is isolated by checking if RCS pressure is rising, which is typically indicative of leak isolation.
d. **INCORRECT** – Steps 1 through 4 of EPP-20 direct the operators to isolate connections between RHR and the RCS. Plausible, since EPP-20 does direct the operators to isolate RCP seal return flow in Step 6. As different leak locations are isolated throughout EPP-20, the operators determine whether or not the leak is isolated by checking if RCS pressure is rising, which is typically indicative of leak isolation. The second part of the distractor is plausible since, per the basis document of EPP-20, certain leak sizes and locations may require more time for RCS re-pressurization to take place after leak isolation, therefore other means of verifying break isolation should be checked (i.e. PZR level). However, the procedure steps in EPP-20 do not direct the operators to use any indications other than RCS pressure.

Technical Reference(s): EPP-20, Revision 9 “LOCA Outside Containment”
EPP-20-BD, Revision 9 “EPP-20 Basis Document”

Proposed references to be provided to applicants during examination: none

Question Source: 2012 Farley Exam

Question History: 2012 Farley Exam

Question Cognitive Level: Fundamental knowledge or memory

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant must demonstrate knowledge of the interrelationship between a LOCA outside containment and systems that connect to the RCS with regard to leak identification and isolation per the guidance in EPP-20.

QUESTION 18

WE05EK1.3

K/A number and description:

WE05EK1.3: Westinghouse. Knowledge of the operational implications of the following concepts as they apply to the Loss of Secondary Heat Sink: Annunciators and conditions indicating signals, and remedial actions associated with the Loss of Secondary Heat Sink.

Question:

A reactor trip has occurred due to a loss of all feedwater. The following conditions exist:

- The crew has recently entered 1-FR-H.1, Response to Loss of Secondary Heat Sink from E-1.
- Annunciators AFW PMP A LO DISCH PRESS/MTR TRIP, AFW PMP B LO DISCH PRESS/MTR TRIP and SD AFW PMP LO DISCH PRESS TRIP are all lit.
- SG wide range levels are: A: 8 % B: 15 % C: 16 %
- All SG pressures are 1040 psig
- RCS pressure is 2300 psig
- Containment pressure is 3 psig
- CST level is 5% and lowering due to sabotage and remains inaccessible

In accordance with FRP-H.1, as a result of the low CST level, SW backup _____(1)_____ required to be aligned to the SD AFW pump.

After action has been taken to address the CST level, _____(2)_____ is/are performed NEXT.

	(1)	(2)
A.	is	attempts to establish feedwater flow
B.	is NOT	attempts to establish feedwater flow
C.	is	stopping all RCPs
D.	is NOT	stopping all RCPs

Distractor Analysis:

A. Incorrect. The first part is incorrect per FR-H.1 step 6 RNO, the SW backup is only aligned to the MDAFW per Attachment 2 due to the catastrophic failure and inaccessibility of the CST. Plausible because if the CST low level was not due to catastrophic failure and was accessible then OP-402 would be used to align SW backup to all AFW pumps. The second part is incorrect. After addressing the CST at step 6 RNO, FR-H.1 directs tripping the RCPs at step 15 prior to attempting the establishment of feedwater. Plausible because the RCP trip criteria of step 4 is not met and attempts to establish feedwater will immediately follow trip of RCPs at step 15.

B. Incorrect. The first part is correct. Per FR-H.1 step 6 RNO, the SW backup is only aligned to the MDAFW per Attachment 2 due to the catastrophic failure and inaccessibility of the CST. The second part is incorrect. After addressing the CST at step 6 RNO, FR-H.1 directs tripping the RCPs at step 15 prior to attempting the establishment of feedwater. Plausible because the RCP trip criteria of step 4 is not met and attempts to establish feedwater will immediately follow trip of RCPs at step 15.

C. Incorrect. The first part is incorrect per FR-H.1 step 6 RNO, the SW backup is only aligned to the MDAFW per Attachment 2 due to the catastrophic failure and inaccessibility of the CST. Plausible because if the CST level was not due to catastrophic failure and was accessible then OP-402 would be used to align SW backup to all AFW pumps. The second part is correct. After addressing the CST at step 6 RNO, FR-H.1 directs tripping the RCPs at step 15 prior to attempting the establishment of feedwater.

D. CORRECT. The first part is correct. Per FR-H.1 step 6 RNO, the SW backup is only aligned to the MDAFW per Attachment 2 due to the catastrophic failure and inaccessibility of the CST. The second part is correct. After addressing the CST at step 6 RNO, FR-H.1 directs tripping the RCPs at step 15 prior to attempting the establishment of feedwater.

Reference(s)

- FRP -H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK, R27
- FRP-H.1 Bases
- APP-7, CONDENSATE AND FEEDWATER, R42

K/A Match discussion:

The question requires knowledge of operational implications of annunciators associated with the loss of Secondary Heat Sink and remedial actions as specified in FR-H.1.

Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis __X___

Question Source: Modified Bank: North Anna 2006-302, RO retake, Q 59. Changed to a 2x2 question due to non-plausibility of two answers. Focused on FR-H.1 differences at RNP from the WOG to address low CST level and resultant procedure flow.

SRO Only Basis: N/A

QUESTION 19

003AK3.05

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	003AK3.05
	Importance Rating	3.4

003AK3.05 – Knowledge of the reasons for the Tech Spec limits for reduction of load to 50% power if flux cannot be brought back within specified target band as they apply to the dropped control rod

Proposed Question:

The plant is at 100% power when the requirements of LCO 3.2.4, QPTR, are not satisfied.

Which one of the following states:

(1) the maximum power level at which the LCO will no longer apply, and
(2) the reason for the power level stated in the LCO applicability statement?

	(1)	(2)
a.	15%	prevent unanalyzed xenon effects on power distribution during subsequent power increases
b.	15%	achieve a condition where the power distribution is no longer a significant concern to safety analyses
c.	50%	prevent unanalyzed xenon effects on power distribution during subsequent power increases
d.	50%	achieve a condition where the power distribution is no longer a significant concern to safety analyses

Proposed Answer:

d. 50% achieve a condition where the power distribution is no longer a significant concern to safety analyses

Explanation:
This question requires the applicant to evaluate plant conditions with LCO 3.2.4 requirements for QPTR not met and determine the maximum allowable power level at which the LCO will no longer apply (and the reason for the applicability limit).

a. **INCORRECT** – The LCO 3.2.4 requirements for QPTR are only applicable in Mode 1 with thermal power greater than 50%. The only way to fully prevent unanalyzed xenon effects on power distribution is to limit the accumulated penalty deviation time for AFD. Thus, both part of the distractor are incorrect (and plausible if the applicant confuses the applicability of AFD with that of QPTR).
b. **INCORRECT** – The LCO 3.2.4 requirements for QPTR are only applicable in Mode 1 with thermal power greater than 50%. The second part of the distractor is the correct reason for the 50% thermal power applicability limit for QPTR. Plausible if the applicant confuses the applicability of AFD with that of QPTR.
c. **INCORRECT** – The LCO 3.2.4 requirements for QPTR are only applicable in Mode 1 with thermal power greater than 50%. The only way to fully prevent unanalyzed xenon effects on power distribution is to limit the accumulated penalty deviation time for AFD. Plausible if the applicant confuses the applicability of AFD with that of QPTR.
d. **CORRECT** – The LCO 3.2.4 requirements for QPTR are only applicable in Mode 1 with thermal power greater than 50%. The reason for reducing power is to place the unit in a condition where the combination of stored energy in the fuel and the energy being transferred to the reactor coolant does not require a QPTR limit on distribution of core power, which per the LCO is a power level $\leq$ 50%.

Technical Reference(s): Tech Spec, Amendment 176, LCOs 3.2.3 and 3.2.4
Tech Spec Bases, Revision 0, LCOs 3.2.3 and 3.2.4

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: comprehension or analysis

10 CFR Part 55 Content: 55.41 (5)

K/A match: The K/A is matched since the applicant is required to demonstrate knowledge of the Tech Spec reasons for lowering power to < 50% thermal power due to a violation of QPTR (which could plausibly occur during a control rod drop).

QUESTION 20

005AG2.1.28

K/A number and description:

005AG2.1.28

005	Generic Abnormal Plant Evolution – Inoperable/Stuck Control Rod
G2.1	Conduct of Operations
G2.1.28	Knowledge of the purpose and function of major system components and controls

Question:

Turbine load was reduced from 95 to 90% to support scheduled maintenance. Control Bank D was inserted from 210 to 197 steps during the load reduction for axial flux control. At 90% power, the Reactor Operator immediately reports that two of the Control Bank D rods, D-8 and H-8, IRPI(s) remain at 210 steps and did not appear to move.

- APP-005-E2, ROD CONT SYSTEM URGENT FAILURE – NOT illuminated
- All Power Range NI's indicate 90%

In accordance with AOP-001, Malfunction of Reactor Control System, **UNTIL** the problem is determined to be either an IRPI indicator(s) issue or actual stuck rods, Tave may be maintained on program using ___(1)____ .

SUBSEQUENTLY, I&C Maintenance and Reactor Engineering support determined that the problem with D-8 was an IRPI indication problem and only H-8 is misaligned. All Control Bank D rods except H-8 ___(2)____ be transferred to the Hold Bus to support rod realignment per AOP-001, Section B.

	(1)	(2)
A.	Turbine load adjustment	will
B.	Turbine load adjustment	will NOT
C.	Control Bank D	will
D.	Control Bank D	will NOT

Proposed Answer: B

Distractor Analysis:

A. Incorrect. First part is correct; either turbine load is adjusted or boron concentration changes are used to maintain Tave on program. AOP-001 requires the operator to assume a rod is misaligned in the absence of indications that the IRPI system is a problem and thus not move rods. Depending on core location, core flux patterns may not be sufficiently abnormal to indicate a rod alignment problem on the ex-core detectors. Second part is incorrect; plausible because the Hold Bus is designed to hold a group of rods to support maintenance; however, with a misaligned rod, the Lift Coil Disconnect Switches are used to support rod realignment.

B. Correct. First part is correct; either turbine load is adjusted or boron concentration changes are used to maintain Tave on program. AOP-001 requires the operator to assume a rod is misaligned in the absence of indications that the IRPI system is a problem and thus not move rods. Depending on core location, core flux patterns may not be sufficiently abnormal to indicate a rod alignment problem on the ex-core detectors. The second part is correct; with a misaligned rod, the Lift Coil Disconnect Switches are used to support rod realignment, NOT the Hold Bus.

C. Incorrect. The first part is incorrect; either turbine load is adjusted or boron concentration changes are used to maintain Tave on program. AOP-001 requires the operator to assume a rod is misaligned in the absence of indications that the IRPI system is a problem and thus not move rods. Plausible because the applicant may consider the problem to be an IRPI indication issue because the PRNIs match and thus determine rods are available. The second part is incorrect; plausible because the Hold Bus is designed to hold a group of rods to support maintenance; however, with a misaligned rod, the Lift Coil Disconnect Switches are used to support rod realignment.

D. Incorrect. The first part is incorrect; either turbine load is adjusted or boron concentration changes are used to maintain Tave on program. AOP-001 requires the operator to assume a rod is misaligned in the absence of indications that the IRPI system is a problem and thus not move rods. Plausible because the applicant may consider the problem to be an IRPI indication issue because the PRNIs match and thus determine rods are available. The second part is correct; with a misaligned rod, the Lift Coil Disconnect Switches are used to support rod realignment, NOT the Hold Bus.

Reference(s)

- AOP-001, Malfunction of Reactor Control System
- AOP-001-BD, Malfunction of Reactor Control System Basis Document
- System Description, SD-007, Rod Control System, Rev 8

K/A Match discussion:

The K/A is matched because the question involves knowledge of the ESFAS system purpose.

Cognitive Level:	Comprehension or Analysis	__X_
Question Source: New:	___X__	

QUESTION 21

024AA2.02
Emergency Boration /1
Ability to determine and interpret the following as they apply to the Emergency Boration: When use of the manual boration valve is needed
RO 3.9/SRO 4.4
CFR 43.5

Initial Conditions:

- Unit 2 was at 100% power
- A manual reactor trip was unsuccessfully attempted due to a Feedwater Controller malfunction
- The crew has entered FRP-S.1, "Response to Nuclear Power Generation/ATWS"

Current Conditions:

- Operators have just reached the step in FRP-S.1 to initiate Emergency Boration

To maximize borated water flow to the RCS, FRP-S.1 states that the HIC-121 potentiometer, "Charging Controller – DEMAND SIGNAL" is to be established at ____(1)____ .

Per FRP-S.1, ____(2)____ is the preferred, highest flowrate, Charging Pump supply source.

	(1)	(2)
A.	0 %	MOV-350, BA TO CHARGING PMP SUCT
B.	0 %	CVC-358, RWST TO CHARGING PUMP SUCTION
C.	100 %	MOV-350, BA TO CHARGING PMP SUCT
D.	100 %	CVC-358, RWST TO CHARGING PUMP SUCTION

Distractor Analysis

A. Correct. Given information provides applicant success path.

B. Incorrect. Answer is plausible due to being partially correct. CVC-358 is only used in FRP-S.1 if Boric Acid pump supply or MOV-350 are precluded from use. In addition, CVC-358 provides a charging pump suction source with less driving head than one using MOV-350 with an associated BA pump supply. (Ref FRP-S.1, Page 6 and FRP-S.1-BD, Page 49)

C. Incorrect. Answer is plausible due to being partially correct. The HIC-121 controller set to 100% drives the controlled HCV-121 closed, preventing Charging Pump supply from establishing maximum charging flow. (Ref FRP-S.1, Page 5)

D. Incorrect. Plausible if applicant misapplies incorrect answers to question, see above.

References

- FRP-S.1, Revision 20, Pages 5 & 6
- FRP-S.1-BD, Revision 20, Page 49

KA Match

Applicant is supplied conditions where system knowledge must be applied to plant conditions to derive correct answer.

Cognitive Level

Low, due to systems knowledge necessary to arrive at correct conclusion.

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 22

028AA1.02

APE028 AA1.02
Pressurizer Level Control Malfunction
Ability to operate and/or monitor the following as they apply to Pressurizer Level Control Malfunctions: CVCS 3.4 /3.4 (41.7, 45.5, 45.6)

Given the following:

- The plant is operating at 100% RTP.
- 'A' Charging Pump is in Auto and its speed is lowering.
- Pressurizer Backup Group 'A' heaters are in AUTO and have energized.
- Backup Group 'B' heaters are in MANUAL

Based on these indications, the reference leg for **__(1)__** is leaking.
With no operator action, a letdown isolation signal will eventually occur.
Following the isolation, Backup Group 'B' heaters will be **__(2)__**.

	(1)	(2)
A.	LT-459	energized
B.	LT-460	energized
C.	LT-459	deenergized
D.	LT-460	deenergized

Distractor Analysis

A leak in the reference leg of a level transmitter causes its pressure to lower, causing a larger △P across the level transmitter, which looks like higher level. The pressurizer level control system sees this as PZR level rising, so it lowers charging flow to attempt to compensate.

LT-459 indication is failing high, but actual PZR level as measured by LT-460 is lowering, so eventually 460 will isolate letdown.

RNP normally operates with Control Group heaters and one set of Backup heaters in Auto. The other set of Backup heaters is in Manual to provide continuous spray to maintain boron homogeneity between the RCS and PZR.

A.. Incorrect.

B. Incorrect.

C. **CORRECT**. (1) Only LT-459 will energize backup group heaters when PZR level is 5% above program (LT-460 doesn't do this), so 459 is the correct choice.
(2) Only LT-460 deenergizes ALL PZR heaters at 14.4% (459 only deenergizes those in AUTO). After letdown isolates, continued minimum charging flow and seal injection flow will cause PZR level to increase (assuming the break is small enough). Letdown will automatically **un**isolate, but PZR heaters must be manually reset.

D. Incorrect.

References:
ST-059, Pressurizer/PRT System, Revision 3

KA Match:
The KA is matched because the applicant must be able to diagnose a Pressurizer Level Control system malfunction and predict a change in the CVCS system.

Cognitive Level: **High**
Higher cognitive level as the KA requires prediction of an impact based on analysis of given conditions.

Source of Question: RNP Master Bank, question 122, modified. Original question only tested knowledge of which transmitter was failing. This question adds a letdown isolation/heater cutoff concept.

QUESTION 23

032AK1.01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	AK1.01	
	Importance Rating	2.5	

Proposed Question:

0800 The reactor is shut down in Mode 6 and refueling of the core is in progress. All Source Range Nuclear Instrument channels and Post Accident Monitor Source Range channels are operable.

0810 Source Range Nuclear Instrument channel N-31 power supply failure causes detector supply voltage to slowly lower by approximately 200 volts.

0820 N-31 power supply output is 0 volts and N-31 has been declared inoperable.

At 0810 Control room indication of N-31 will be ___(1)___ as compared to N-32.
At 0820 In accordance with Tech Spec 3.9.2, core alterations __(2)___ continue.

Which one of the following completes the statements above?

A	(1) the same (2) may
B	(1) the same (2) may NOT
C	(1) lower (2) may
D	(1) lower (2) may NOT

Proposed Answer: **C**

K/A Match Analysis

032 Loss of Source Range Nuclear Instrumentation, AK1.01

Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: Effects of voltage changes on performance.

Requires knowledge of the effect of a voltage change on the performance of source range nuclear instrument N-31.

Answer Choice Analysis

A. (1) INCORRECT. Plausible because the IR outer chamber and the PR detectors operate in the ionization region of the gas amplification curve for gas filled detectors and a detector that operates in this region will have the same output with small changes in detector voltage.
   (2) CORRECT. Tech Spec 3.9.2 requires the verification of one PAM SR monitor within 15 minutes if one of two required SR monitors is inoperable.
B. (1) INCORRECT.
   (2) INCORRECT. Plausible because Tech Spec 3.9.2 requires the suspension of core alterations if two of two required SR monitors are inoperable, or if required actions/completion times of condition A are not met.
C. (1) CORRECT. The SR detectors operate in the Proportional region of the gas amplification curve and the amplification effect will decrease with decreasing voltage in the Proportional region.
   (2) CORRECT.
D. (1) CORRECT. (2) INCORRECT.

Technical Reference(s):

Tech Spec 3.9.2 amendment 190

Tech Spec 3.9.2 Basis rev. 39

FMP-019 Fuel Insert and Shuffle, rev. 41, Attachment 10.5 Guidelines for Performing 1/M Plots During Core Loading

Proposed references to be provided to applicants during examination: None

Learning Objective:

FMP-019-LP rev 0

Objective 6: STATE the Technical Specification requirements associated with FMP-019 and EXPLAIN their bases.

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

Attached

QUESTION 24

033AA2.09

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	033AA2.09
	Importance Rating	3.4

033AA2.09 – Ability to determine and interpret the conditions which allow bypass of an intermediate range level trip switch as they apply to the Loss of Intermediate Range Nuclear Instrumentation

Proposed Question:

Power ascension is in progress during a plant startup with the following power range nuclear instrument readings:

- N-41 = 10.5%
- N-42 = 10.5%
- N-43 = 9.0%
- N-44 = 9.5%

Intermediate range nuclear instrument (N-35) has been declared inoperable due to a loss of its compensating voltage. Per APP-005, the Unit Supervisor has directed you to bypass N-35 in accordance with OWP-011 "Nuclear Instrumentation."

You have just placed the intermediate range level trip switch for N-35 in the BYPASS position.

Given the above plant conditions, which one of the following describes the status of LCO 3.3.1, "RPS Instrumentation," AND the actions (if any) that are needed to comply with LCO 3.3.1?

a. No further action is necessary. The IRNI neutron flux function of LCO 3.3.1 is NOT applicable in the current plant conditions.

b. The IRNI neutron flux function of LCO 3.3.1 is applicable in the current plant conditions and the conditions of the LCO are met.

c. LCO 3.3.1 is NOT met; power must be lowered below P-6.

d. LCO 3.3.1 is NOT met; power must be raised above P-10.

Proposed Answer:

a. No further action is necessary. The IRNI neutron flux function of LCO 3.3.1.1 is NOT applicable in the current plant conditions.

Explanation:

This question requires the applicant to evaluate plant conditions with an inoperable IRNI and make the determination that, based on two PRNI power levels being at 10%, the P-10 permissive logic is satisfied. The applicant must then use knowledge of LCO 3.3.1 applicability to correctly determine that with one IRNI channel inoperable, taking the affected instrument to bypass does not require any further Tech Spec required action (no IRNI channels are required if less than P-6 OR greater than P-10).

a. **CORRECT** – Since reactor power is greater than P-10, N-35 can be bypassed without taking any further Tech Spec required actions.
b. **INCORRECT** – Since reactor power is greater than P-6 and greater than P-10, the IRNI neutron flux function of LCO 3.3.1 is not applicable. Plausible if the applicant determines that the P-10 permissive logic is not met with only 2 PRNI's at 10% power.
c. **INCORRECT** – Since reactor power is greater than P-6 and greater than P-10, the IRNI neutron flux function of LCO 3.3.1 is not applicable. Plausible if the applicant determines that the P-10 permissive logic is not met with only 2 PRNI's at 10% power.
d. **INCORRECT** – Since reactor power is greater than P-6 and greater than P-10, the IRNI neutron flux function of LCO 3.3.1 is not applicable. Plausible if the applicant determines that the P-10 permissive logic is not met with only 2 PRNI's at 10% power.

Technical Reference(s): Tech Spec, LCO 3.3.1 "RPS Instrumentation"
SD-010, Revision 9 "Nuclear Instrumentation System"
APP-005
OWP-011 "Nuclear Instrumentation"
(see attached markups)

Proposed references to be provided to applicants during examination: none

Question Source: new

Question Cognitive Level: comprehension or analysis

10 CFR Part 55 Content: 55.41 (10)

K/A match: The K/A is matched since the applicant is expected to evaluate plant conditions during a loss of an intermediate range nuclear instrument and, based on those plant conditions, determine the Tech Spec conditions which allow bypassing the failed instrument (by taking the level trip switch on the failed instrument to bypass).

QUESTION 25

061AA1.01

K/A number and description:
061AA1.01: Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Automatic actuation

Question:

Given the following plant conditions:

- Unit at 100% power.
- Control Room Air Cleaning fan, HVE-19A, is tagged out for scheduled motor inspection.
- A valid high radiation alarm on R-1 occurs

Complete the following statement:

Control room exhaust fan, HVE-16, is shutdown ___(1)___ and Aux Building Supply Fan, HSV-1 ___(2)___ required to be shutdown.

	(1)	(2)
A.	automatically	is
B.	automatically	is NOT
C.	manually	is
D.	manually	is NOT

Proposed Answer: A

Distractor Analysis:

A. Correct. Either Fan 19A or 19B receiving an auto stop will cause the exhaust fan to auto stop. The AB supply fan is required to be shutdown manually with a valid R-1 actuation.

B. Incorrect. Part (1) is correct; either Fan 19A or 19B receiving an auto stop will cause the exhaust fan to auto stop. Part (2) is incorrect. The answer is plausible because the action is not directed under the APP-036-D7 for “Area Monitor Hi Rad” but under a separate APP-10-A8 for the “CR Fan Trouble/Rad Start” and the applicant may not be aware of the separate APP action.

C. Incorrect. Part (1) is incorrect; the answer is plausible because a note in OP-906 describes the logic of either HVE-19A or HVE-19B starting from a high radiation signal will de-energize HVE-16. The applicant may incorrectly assume the logic requires both 19A and 19B to start or that the fact that 19A being tagged may impact the auto actions. Part (2) is correct. The AB supply fan is required to be shutdown manually with a valid R-1 actuation.

D. Incorrect. Part (1) is incorrect; the answer is plausible because a note in OP-906 describes the logic of either HVE-19A or HVE-19B starting from a high radiation signal will de-energize HVE-16. The applicant may incorrectly assume the logic requires both 19A and 19B to start or that the fact that 19A being tagged may impact the auto actions. Part (2) is incorrect. The answer is plausible because the action is not directed under the APP-036-D7 for “Area Monitor Hi Rad” but under a separate APP-10-A8 for the “CR Fan Trouble/Rad Start” and the applicant may not be aware of the separate APP action.

Reference(s)

- APP-036-D7, Alarm response for AREA MONITOR HI RAD
- APP-010-A8, Alarm response for CR FAN TROUBLE/RAD START
- OP-905
- AOP-005

K/A Match discussion:

This question meets the K/A because the candidate must determine if auto actions associated with an area rad monitor valid alarm occur for the given plant conditions.

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

Question Source: Bank: _______
Modified Bank: _______ (Note changes or attach parent)
New: ___X___

QUESTION 26

067AK3.02

Plant Fire On-Site /8
Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Steps called out in the site fire protection plan, FPS manual and fire zone manual
RO 2.5/SRO 3.3
CFR 41.5/41.10

Initial Conditions:

- Unit 2 is currently operating at 100% RTP
- A fire has been reported in a LHRA

Complete the following statements in accordance with FP-001, "Fire Emergency".

Prior to authorizing entry into this fire area, a Radiological Controls (RC) brief ____(1)____ required.

If a RESCUE situation develops, establishment of a dedicated Back-up Team ____(2)____ required.

	(1)	(2)
A.	is still	is
B.	is NOT	is
C.	is Still	is NOT
D.	is NOT	is NOT

Distractor Analysis

A. Incorrect. Answer is plausible due to being partially correct. Establishment of a Back-up Team is not required during a rescue situation. (Ref FP-001, page 39)

B. Incorrect. Plausible if applicant misapplies incorrect answers to question, see "A" and "D" distractor analyses.

C. Correct. Given information provides applicant success path.

D. Incorrect. Answer is plausible due to being partially correct. An RC brief is required when conducting fire-fighting activities in LHRA's. (Ref FP-001, page 18)

References

- FP-001, Revision 64, Pages 18 and 39

KA Match

Applicant is given a situation where duties of a Fire Brigade Incident Commander (FBIC) overlap supplied fire conditions.

Cognitive Level

Low

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 27

WE09EK2.2

WE09 EK2.2
Natural Circulation Operations
Knowledge of the interrelations between Natural Circulation Operations and: Facility's heat removal systems, including primary coolant, emergency coolant, decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.
3.6/3.9 (41.7, 45.7)

Given the following:

- A loss of off-site power occurred 15 minutes ago.
- The crew is performing EOP-ES-0.1, Reactor Trip Response.
- The BOP is verifying Natural Circulation in accordance with ES-0.1 Attachment 1, Natural Circulation Verification:
  - RCS pressure is 1800 psig and stable.
  - SG pressures are all 1000 psig and stable.
  - RCS Thot is 586^{o}F and trending down.
  - RCS Tcold is 547^{o}F and trending down.
  - Core Exit Thermocouple temperature is 591^{o}F and trending down.

The Natural Circulation criteria in Attachment 1 __**(1)**__ met.

Over the next 60 minutes, RCS Core Delta T will __**(2)**__.

	(1)	(2)
A.	are	lower
B.	are	rise
C.	are NOT	lower
D.	are NOT	rise

Distractor Analysis

A. Incorrect. 1st part incorrect, as discussed in C. Plausible if the applicant chooses T_{HOT} as the highest RCS temperature, in which case Subcooling would be 36F so that criterion would be met. 2nd part correct as discussed in C.

B. Incorrect. 1st part incorrect as discussed above. 2nd part is incorrect but plausible because initially core delta T increases, but then natural circulation flow is developed and $\triangle$T begins to lower as decay heat is removed from the core. The conditions given make it clear that natural circ flow is established.

C. CORRECT. Att. 1 Natural Circ criteria: Subcooling >35F; S/G pressures, T_{HOT}, & core exit TCs all stable or lowering; T_{COLD} at saturation temp for S/G pressure.

- T_{SAT} for 1800 psig (1814.7 psia) is 622.3F. With highest RCS temp at 591F, subcooling is 31F, so Subcooling is **NOT** met.
- S/G pressures are stable
- T_{HOT}, & core exit TCs are lowering
- Saturation temp for S/G pressure (1014.7 psia) is 546.5F

Core delta T lowers as core decay heat lowers over time.

D.. Incorrect. 1st part correct as discussed in C. 2nd part incorrect as discussed in B.
Handouts Provided: Steam Tables

References:
EOP-ES-0.1, Reactor Trip Response

KA Match:
The KA is matched because applicants must demonstrate knowledge of the interrelations between natural circulation and the facility's heat removal systems (primary coolant and steam generators) and relations between proper operation of these systems to operation of the facility.

Cognitive Level: **High**

Source of Question: Robinson exam bank Question 461. Changed format to 2x2, but didn't change content. Updated procedure names.

QUESTION 28

003K5.04

003K5.04
Reactor Coolant Pump
Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow
RO 3.2/SRO 3.5
CFR 41.5

Current Conditions:

- Unit 2 was at 35% power, End of Life (EOL) following 300 days of continuous operation.
- A bus fault has resulted in loss of 4kV Bus 4

Based on the current conditions, RCP B has tripped on ____(1)____ .

Two minutes following the loss of RCP B, SG B steam flow is ____(2)____ than SG's A and C.

	(1)	(2)
A.	under-frequency	higher
B.	under-frequency	lower
C.	under-voltage	higher
D.	under-voltage	lower

Distractor Analysis

A. Incorrect. Plausible if applicant misapplies incorrect answers to question, see "B" and "C" distractor analyses.

B. Incorrect. Answer is plausible due to being partially correct. The RCP under-frequency trip requires two out of three RCP Bus under-frequencies to trip all RCP's. The given information provides only a single RCP trip and is therefore indicative of an under-voltage trip. (Ref ST-001, page 35)

C. Incorrect. Answer is plausible due to being partially correct. Steam Flow will lower due to the lower generator pressure environment, SG's A and C will make up for the lack of SG B steam flow.

D. Correct. Given information provides applicant success path.

References

- ST-001, Revision 4, Page 35

KA Match

Applicant is given a situation where a diagnosis of plant conditions has resulted in a loss of a RCP and has to predict the corresponding steam generator impact.

Cognitive Level

High

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 29

003K6.04

Question 29
003K6.04
Reactor Coolant Pump
Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation
RO 2.8/SRO 3.1
CFR 41.7

The following sequence of events has occurred on Unit 2:

- 0900: CVC-381, "RCP Seal Return Isolation" is closed and cannot be re-opened
- 0903: Receipt of annunciator APP-001-C1, "RCP THERM BAR COOL WTR HI FLOW"
- 0904: Receipt of annunciator APP-001-D1, "RCP THERM BAR COOL WTR LO FLOW"
- 0905: Unit 2 is at 70% RTP

Based on these conditions, a #1 seal return flow-path is still available to the ____(1)____ , continued RCP operation ____(2)____ permitted.

	(1)	(2)
A.	Reactor Coolant Drain Tank (RCDT)	is
B.	Pressurizer Relief Tank (PRT)	is
C.	Reactor Coolant Drain Tank (RCDT)	is NOT
D.	Pressurizer Relief Tank (PRT)	is NOT

Distractor Analysis

A. Incorrect. Answer is plausible due to being partially correct. CVC-382, "RCP Seal Return Line Relief" will lift to permit continued seal operation with CVC-381 closed. CVC-382 discharges to the PRT. (Ref DWG 5379-685, Sheet 1)

B. Correct. Given information provides applicant success path.

C. Incorrect. Plausible if applicant misapplies incorrect answers to question, see "A" and "D" distractor analyses.

D. Incorrect. Answer is plausible due to being partially correct. Per AOP-14-BD, RCP operation may continue in the event of a Thermal Barrier failure as long as Seal Injection is maintained. (Ref AOP-14-BD, Page 18)

References

- DWG 5379-685, Sheet 1
- APP-001, Revision 57, Pages 24, 25, 33, 34
- ST-001, Revision 4, Page 43
- AOP-014-BD, Revision 36, Page 18

KA Match

Applicant is given a situation where a diagnosis of plant conditions and systems knowledge is required to determine RCP impact due to a containment isolation valve failure.

Cognitive Level

Low

Source of Question

Modified, 2009-301 NRC Exam

SRO Only Basis

Not applicable

QUESTION 30

004K3.06

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	004K3.06
	Importance Rating	3.4

004K3.06 – Knowledge of the effect that a loss or malfunction of the CVCS will have on RCS temperature and pressure.

Proposed Question:

Given the following:

- The plant is operating at 100% power with CVCS in its normal letdown alignment when the Letdown Temperature element (TE-144) fails upscale.
- Pressurizer Pressure Control is in MANUAL
- Rod Control is in MANUAL

Based on the conditions above, VCT/Demineralizer Diversion Valve (TCV-143) will ___(1)___ and RCS temperature and pressure will ___(2)___.

	(1)	(2)
a.	bypass the demineralizers	remain unchanged
b.	bypass the demineralizers	decrease
c.	maintain its current position	increase
d.	maintain its current position	decrease

Proposed Answer:

c. maintain its current position increase

Explanation:

This question requires the applicant to recognize the effect of a failed temperature element (TE-144) on CVCS as well as the effects that will be seen in RCS temperature and pressure. The applicant should recognize that the upscale failure of the temperature element will cause maximum CCW cooling flow to the non-regenerative heat exchanger (NRHX). Overcooling the resin in the mixed bed demineralizers will cause an overall decrease in boron concentration which will cause reactor power to increase, thus increasing RCS temperature and pressure. The question stem asks for the overall effects assuming no automatic control system response (i.e. rod control or pressurizer pressure control).

a. **INCORRECT** – TCV-143 receives its input from TE-143 (not TE-144), thus this failure will not result in bypassing the demineralizers; if the applicant incorrectly determines that TCV-143 repositions due to high temperature, then there would not be an expected change in boron concentration leading to changes in RCS temperature and pressure; plausible if the applicant incorrectly assumes the failed temperature element leads to less CCW flow to the NRHX
b. **INCORRECT** – TCV-143 receives its input from TE-143 (not TE-144), thus this failure will not result in bypassing the demineralizers; if the applicant incorrectly determines that TCV-143 repositions due to high temperature, then there could not be an expected change in boron concentration leading to decreasing RCS temperature and pressure (as temperature rises up to the point where the valve repositions, overheating the resins could lead to an increase in boron concentration to the reactor); plausible if the applicant incorrectly assumes the failed temperature element leads to less CCW flow to the NRHX
c. **CORRECT** – TCV-143 receives its input from TE-143 (not TE-144), thus this failure will not result in bypassing the demineralizers; colder water entering the demineralizers due to maximum CCW flow to the NRHX will cause an overall boron concentration decrease; thus RCS temperature and pressure will increase
d. **INCORRECT** – TCV-143 receives its input from TE-143 (not TE-144), thus this failure will not result in bypassing the demineralizers; if the applicant incorrectly determines that the colder water entering the demineralizers causes overall boron concentration to increase then RCS temperature and pressure would be expected to decrease

Technical Reference(s): SD-021, Revision 12 “Chemical and Volume Control System” (see attached markups)

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: N/A

Question Cognitive Level: comprehension or analysis

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant is expected to evaluate the effects of a failure of a temperature element in the CVCS and determine the impacts of the failure on RCS temperature and pressure.

QUESTION 31

005K4.12

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K4.12	
	Importance Rating	3.1	

Proposed Question:

Given the following plant conditions:
- A Large Break LOCA and Loss of Offsite Power have occurred.
- EDG B tripped while starting.
- RWST level is 9%.
- Alignment to the CV Sump has been completed.
- CV Pressure is currently 12 psig.

SI-844A and B, CV Spray Pump Suction Isolation Valves, will be __(1)__and RHR Pump A is capable of supplying suction to CV Spray Pump(s) __(2)__.

Which one of the following completes the statement above?

A	(1) closed (2) "A" ONLY
B	(1) closed (2) "A" and "B"
C	(1) open (2) "A" ONLY
D	(1) open (2) "A" and "B"

Proposed Answer: **D**

K/A Match Analysis
005 Residual Heat Removal System (RHRS), K4.12
Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Lineup for piggyback mode with CSS.

Requires knowledge of design features which allow for piggyback of CS onto the RHR system.

Answer Choice Analysis

A. Incorrect - SI-844A/B are normally open valves and are not impacted by the loss of

E2. SI-844A/B are not cycled prior to or during alignment for piggy-back mode of operation. Without power to "B" train components only "A" CV Spray Pump will be operable. Also, Attachment 1 of EPP-9 closes RHR Pump Suction Valves RHR-752A/B prior to initiating cold leg recirculation. The candidate could confuse this with having to close SI-844s. The candidate may think that RHR Pump "A" cannot be aligned to supply suction to both trains of spray with only E-2 available.

B. Incorrect - SI-844A/B are normally open valves and are not impacted by the loss of E2. SI-844A/B are not cycled prior to or during alignment for piggy-back mode of operation. The second part of the distractor is correct.

C. Incorrect - The first part of the distractor is correct. The candidate may think that RHR Pump "A" cannot be aligned to supply suction to both trains of spray with only E-2 available.

D. Correct - SI-844A/B are not cycled prior to or during alignment for piggy-back mode of operation. RHR Pump "A" can still be aligned to supply suction pressure to both "A" and "B" CV Spray Pump.

Technical Reference(s):

EPP-9, TRANSFER TO COLD LEG RECIRCULATION, Rev 35

ST-024, CONTAINMENT SPRAY SYSTEM, Rev 2a

Proposed references to be provided to applicants during examination: None

Learning Objective:

CSS SD-024-009

Question Source:	Robinson 2011-302 exam Q.42	
Question History:	Last NRC Exam: Robinson 2011-302	
Question Cognitive Level:	Memory or Fundamental Knowledge	_
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 X	
	55.43 _	

Comments:

Supporting References

Attached

QUESTION 32

005K5.05

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	K5.05	
	Importance Rating	2.7	

Proposed Question:

Unit 2 plant conditions:

- Plant startup activities following a refueling outage are in progress.
- The Pressurizer is solid.
- RCP 'A' is running.
- PCV-145, LOW PRESSURE LETDOWN VALVE, has been placed in MANUAL.
- GP-002 (Cold Shutdown To Hot Subcritical At No Load Tavg) Section 8.2 (Instructions for Plant Heatup up to 199ºF) is in progress.
- RCS temperature is 143º F and increasing.

As the RCS heats up the operators will adjust PCV-145 by pushing the ___(1)____ button to control RCS pressure.

The operators will throttle ___(2)____ to control RCS heat up rate.

A	down HCV-758, RHR HX DISCH FLOW
B	down FCV-605, RHR HX BYPASS VALVE
C	up HCV-758, RHR HX DISCH FLOW
D	up FCV-605, RHR HX BYPASS VALVE

Proposed Answer: **A**

K/A Match Analysis

005 Residual Heat Removal System (RHRS), K5.05

Knowledge of the operational implications of the following concepts as they apply the RHRS: Plant response during "solid plant": pressure change due to the relative incompressibility of water.

Requires knowledge of the operational implications of the incompressibility of water and the plant response to a pressure change caused by a temperature change caused by the operation of the RHR system.

Answer Choice Analysis
A. CORRECT. The first part is correct; the operators must push the down button on PCV-145 controller to cause the controller to deliver less control air to the valve which in turn will cause PCV-145 to move in the open direction and maintain RCS pressure constant due to thermal expansion of the RCS volume.
The second part is correct; GP-002 section 8.2 contains the procedure steps to adjust HCV-758 to increase RCS temperature.

B. INCORRECT. The first part is correct. The second part is incorrect but plausible; HCV-605 is initially adjusted to maintain a minimum of 3000 gpm and placed in automatic. When RHR flow changes due to HCV-758 being adjusted to initiate heat up, HCV-605 is expected to automatically control total RHR flow at set point.

C. INCORRECT. The first part is incorrect; it is plausible since PCV-145 is air close, spring opening and the operation of the PCV-145 controller in manual requires knowledge of controller demand relative to the air signal sent to the valve in response to increasing RCS pressure or fails to understand the pressure implications of thermal expansion when solid due to the incompressibility of water. The down pushbutton will lower RCS pressure; a common misconception is that the down pushbutton will close the valve. The second part is correct.

D. INCORRECT. The first and second parts are both incorrect.

Technical Reference(s):

GP-001 RCS FILL AND VENT, Rev. 66
GP-002 COLD SHUTDOWN TO HOT SUBCRITICAL AT NO LOAD Tavg, Rev. 126
ST-021 CHEMICAL AND VOLUME CONTROL SYSTEM, Rev. 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Lesson plan LP-21-3 CVCS-Letdown and Charging rev 2, slide 70, Objective 7d.

Question Source: NEW

Question History: NEW

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 33

006A4.02

K/A number and description:

006A4.02: 006: Emergency Core Cooling System (ECCS)
A4: Ability to manually operate and/or monitor in the control room:
A4.02: Valves

Question:

Given the following plant conditions:

- The reactor is at 100% RTP
- A Safety Injection signal is received

PCV-1922 A/B, CONTAINMENT ISOLATION VALVE SEAL WATER (IVSW)
RHR-744 A/B, LOOP DISCHARGE TO RCS ISOLATION VALVES
V12-12/13, VACUUM RELIEF VALVES
SI-867 A/B, BIT INJECTION TANK INLETS

Which of the following valves will **NOT** reposition automatically due the above conditions?

A. PCV-1922 A/B

B. V12-12/13

C. SI-867 A/B

D. RHR-744 A/B

Distractor Analysis:

Proposed Answer: C

Explanation.

A. Incorrect. PCV-1922 A/B will automatically reposition to open due to the ESF Phase A as a result of the SI signal. Plausible because some containment isolations occur as a result of other signals such as RCP cooling water with a Phase B and applicants are required to know differences.

B. Incorrect. V12-12/13 will reposition due to a Containment Vent Isolation signal as a result of the SI. Plausible because some containment isolations occur as a result of other signals such as RCP cooling water with a Phase B and applicants are required to know differences.

C. Correct. SI-867 A/B receive an SI signal to reposition; however, they do not reposition because of existing alignment.

D. Incorrect. RHR 744 A/B will auto open due an SI signal. Plausible because some valves receive will not reposition even with an SI signal due to existing alignment.

Reference(s)

- Student Text ST-006, ESF
- System Description SD-002, Safety Injection

K/A Match discussion:

This question meets the KIA because the K/A met because the applicant must know what valves will reposition for a Safety Injection Signal so they can monitor on the RTGB.

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

Question Source:	Bank:	___	
	Modified Bank:	X	(Note changes or attach parent)
	New:	___	

Modified NRC exam RNS 2013-301, question 33 by changing two distractors which contained the SI Accumulator discharge valves, SI-865 A/B/C

SRO Only Basis: N/A

QUESTION 34

007A2.05

K/A number and description:

SYS007 A2.05

007: Pressurizer Relief Tank/Quench Tank System (PRTS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.05: Exceeding PRT high-pressure limits

Question:

Initial Conditions:

The unit is at 100% power.

The station is addressing a gradual rising PRT temperature and level due to seat leakage on a pressurizer safety valve as a mitigating action until the unit is removed from service for repairs.

In accordance with OP-103, Pressurizer Relief Tank Control System, actions to drain the PRT and makeup with primary water _____(1)_____ allowed to be performed simultaneously.

If the leaking pressurizer safety valve subsequently fails open, the PRT is protected from exceeding its design pressure by rupture disks that operate at the minimum pressure of _____(2)______ psig.

	(1)	(2)
A.	are	100
B.	are	120
C.	are NOT	100
D.	are NOT	120

Distractor Analysis:

A. Incorrect. The first part is incorrect. OP-103 checks to ensure the opposite action (drain or refill) is not occurring before starting the desired action. Significant Event Report 7-93 is referenced for an industry event where a PRT was over-pressurized thereby rupturing the disc. Plausible because the plant can physically support the activities simultaneously and the applicant may be unaware of the procedural restriction. The second part is correct.

B. Incorrect. The first part is incorrect. OP-103 checks to ensure the opposite action (drain or refill) is not occurring before starting the desired action. Significant Event Report 7-93 is referenced for an industry event where a PRT was over-pressurized thereby rupturing the disc. Plausible because the plant can physically support the activities simultaneously and the applicant may be unaware of procedural restriction. The second part is incorrect; the value is plausible because 120 is the value associated with the desired PRT temperature limit.

C. CORRECT. The first part is correct. OP-103 checks to ensure the opposite action (drain or refill) is not occurring before starting the desired action. The second part is correct.

D. Incorrect. The first part is correct. OP-103 checks to ensure the opposite action (drain or refill) is not occurring before starting the desired action. . The second part is incorrect; the value is plausible because 120 is the value associated with the desired PRT temperature limit.

Reference(s)

- OP-103, PRESSURIZER RELIEF TANK CONTROL SYSTEM, R21

K/A Match discussion:

The K/A is matched because the question involves conditions related to drawing a bubble in pressurizer. The operational implications of this evolution are tested by demonstrating knowledge of prerequisites for the evolution and related limits in the event of complications.

Cognitive Level: Memory or Fundamental Knowledge __X__

Comprehension or Analysis ______

Question Source: Bank: ______

Modified Bank: ________ (Note changes or attach parent)

New: ___X___

SRO Only Basis: N/A

QUESTION 35

008A1.02

KA 008 A1.02
Component Cooling Water
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature. 2.9/3.1 (41.5/45.5)

- The plant is at 100% power, steady-state
- The controller for TCV-144, NON-REGEN HX OUTLET TEMP CONTROL VALVE, has just been placed in MANUAL due to erratic operation
- Subsequently, a 60 gpm orifice is placed in service, replacing the 45 gpm orifice that had been in service by itself
- No adjustments have been made to TCV-144

Reactor Coolant System temperature will __**(1)**__ due to reactivity effects. If letdown temperature continues to rise, TCV-143, VCT/DEMIN DIVERSION, will divert letdown flow to the VCT at __**(2)**__.

	(1)	(2)
A.	increase	130º F
B.	increase	135º F
C.	decrease	130º F
D.	decrease	135º F

Distractor Analysis

A. Incorrect. The first part is incorrect but plausible because typically as temperature goes up solubility goes up. So the applicant could rationalize that more boron is removed from solution, thus less returns to the RCS and temperature rises due to the positive reactivity effect. The second part is incorrect but plausible because 130F is the temperature at which TCV-143 will automatically reposition back to the demineralizer flowpath.

B. Incorrect. The first part is incorrect as discussed in A. The second part is correct; both the high temperature alarm and TCV-143 diversion to the VCT occur at 135F as sensed by TE-143 in the letdown line.

C. Incorrect. First part is correct. As letdown flow temperature increases, the ion exchange resin releases boron, the negative reactivity effect of which lowers RCS temperature. The second part is incorrect as discussed in A.

D. **CORRECT**. The first part is correct as discussed in C. The second part is incorrect as discussed in B.

References:
ST-021, Chemical and Volume Control System, Rev. 2

KA Match:
The KA is matched because the applicant must predict changes in parameters (RCS boron concentration and temperature) associated with mis-operation of a CCW temperature control valve, and is tested on knowledge of when a protective action occurs to prevent overheating ion exchange resign (to prevent exceeding design limits).

Cognitive Level: Low

Source of Question: McGuire 2010 NRC exam, slightly modified.

QUESTION 36

008K3.03

008K3.03
Component Cooling Water
Knowledge of the effect that a loss or malfunction of the CCWS will have on: RCP
4.1/4.2 (No CFR reference listed.)

Given the following:

- Unit is in Mode 2.
- The crew is performing actions of AOP-014, Component Cooling Water System Malfunction, Section D, due to low CCW System flow and rising temperature.

Any Reactor Coolant Pump whose __(1)__ bearing temperature exceeds __(2)__ must be stopped.

	(1)	(2)
A.	motor	185º F
B.	pump	185º F
C.	motor	200º F
D.	pump	200º F

Correct Answer: C

<u>Distractor Analysis</u>

<u>Support for Correct Answer</u>
AOP-014 has four sections for various CCW malfunctions.
Section D covers "Low Flow <u>OR</u> High Temperature," as given in the stem.
Step 3 is a Continuous Action to "Check ANY RCP **Motor** Bearing Temperature - >200º F."
If Yes, Step 4.a.RNO (Rx not critical) stops all *affected* pumps.
The RCP *motor* bearings (upper & lower) oil systems are cooled directly by CCW.

<u>Plausibility of Incorrect Choices</u>
Section A of AOP-014 covers "Loss of CCW Inventory."
If Step 18 diagnoses a CCW break in containment, Step 19.a will check for the RCP BRG COOL WTR LO FLOW alarm. If Yes, 19.b checks for ANY RCP "BEARING HI TEMP" alarm. (APP-001-B3, D3, & F3 for the individual pump)

These APPs direct you to monitor “RCP Bearing Temperatures” on the “RCP Temperature Recorder and [the] Computer [i.e., ERFIS].” Motor *and* pump bearing points are on the recorder and computer RCP screen. Pump bearing trip setpoint is 185F.

The pump bearings are cooled by seal injection flow, but if that were lost then CCW flow to the thermal barriers provides sufficient cooling. This also lends some plausibility to the *pump* bearing choice.

References:

AOP-014, Component Cooling Water System Malfunction, Rev. 36
AOP-014 Basis Document, Rev. 36
APP-001, Miscellaneous NSSS, Rev. 57

KA Match:

The KA is matched because this malfunction of the CCWS has a direct impact on RCP operation.

Cognitive Level: **Low**

Source of Question: **RNP bank #285.** Converted to 2x2 and the correct answer became C vice D, but content wasn’t changed.

QUESTION 37

010K1.08

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	010K1.08
	Importance Rating	3.2

010K1.08 – Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the PZR LCS.

Proposed Question:

The plant is operating at 100% power with the following conditions:

- Charging pump ‘A’ is running in AUTO
- Charging pump ‘B’ is running in MAN at 20 gpm
- Pressurizer level control switch is selected to 459/460
- ‘A’ Backup Heater Switch is ON
- All other control systems are in AUTO

Pressurizer level transmitter, LT-460, fails downscale.

Which of the following describes the pressurizer spray response and low level alarm status?

	PZR Spray Response	PZR Low Level Alarm Status
a.	No significant change	ON
b.	Increase	ON
c.	No significant change	OFF
d.	Increase	OFF

Proposed Answer:

d. Increase OFF

Explanation:

This question requires the applicant to evaluate given plant conditions in the stem and relate the conditions to a failure of PZR level transmitter (LT-460). The applicant needs to determine that this is the normal backup channel AND that if it fails downscale it will cause the letdown isolation valves to close AND de-energize PZR heaters. Since the manually controlled charging pump is set at 20 gpm; the overall effect of the transmitter failure will be that "actual" PZR level will slowly rise since there will be less letdown flow (0 gpm) than charging flow (~20 gpm as level goes above program level since one of the charging pumps is in MANUAL). As level in the PZR rises, it will compress the steam bubble in the top of the PZR causing pressure to go up to the PZR spray setpoint.

a. **INCORRECT** – When LT-460 fails downscale PZR heaters will de-energize and the letdown isolation valves will close. Since letdown flow will be zero, PZR level will rise ('B' charging pump is in MANUAL), thus compressing the steam bubble in the PZR and making PZR pressure rise to the PZR spray setpoint. The PZR LO LVL alarm comes from the level signal generated by LT-459, so the alarm will not energize based on the failure mode given. Plausible if the applicant determines that the PZR heaters being turned off will arrest any rise in PZR pressure AND if they do not remember that the low level alarm comes from LT-459.
b. **INCORRECT** – When LT-460 fails downscale PZR heaters will de-energize and the letdown isolation valves will close. Since letdown flow will be zero, PZR level will rise ('B' charging pump is in MANUAL), thus compressing the steam bubble in the PZR and making PZR pressure rise to the PZR spray setpoint. The PZR LO LVL alarm comes from the level signal generated by LT-459, so the alarm will not energize based on the failure mode given. Plausible if the applicant does not remember that the low level alarm comes from LT-459.
c. **INCORRECT** – When LT-460 fails downscale PZR heaters will de-energize and the letdown isolation valves will close. Since letdown flow will be zero, PZR level will rise ('B' charging pump is in MANUAL), thus compressing the steam bubble in the PZR and making PZR pressure rise to the PZR spray setpoint. The PZR LO LVL alarm comes from the level signal generated by LT-459, so the alarm will not energize based on the failure mode given. Plausible if the applicant determines that the PZR heaters being turned off will arrest any rise in PZR pressure.
d. **CORRECT** – When LT-460 fails downscale PZR heaters will de-energize and the letdown isolation valves will close. Since letdown flow will be zero, PZR level will rise ('B' charging pump is in MANUAL), thus compressing the steam bubble in the PZR and making PZR pressure rise to the PZR spray setpoint. The PZR LO LVL alarm comes from the level signal generated by LT-459, so the alarm will not energize based on the failure mode given.

Technical Reference(s): SD-059, Revision 9 "Pressurizer System"
SD-021, Revision 12 "Chemical and Volume Control System"
(see attached markups)

Proposed references to be provided to applicants during examination: none

Question Source: 3-Loop Westinghouse Question Bank (see attached parent question)

Question History: VC Summer

Question Cognitive Level: comprehension or analysis

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant is required to assess given plant conditions and determine the effects of a malfunction of a PZR level transmitter on the PZR pressure control system (i.e. PZR heaters).

QUESTION 38

012A2.03

Question 38
012A2.06
Reactor Protection
Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of RPS signal to trip the reactor
RO 4.4/SRO 4.7
CFR 41.5/43.5

MST-021, "Reactor Protection Logic Train "B" at Power", is currently in progress with the following configuration:

- "A" Trip Breaker is racked in and closed
- "A" Bypass Breaker is racked out
- "B" Trip Breaker is racked in and open
- "B" Bypass Breaker is racked in and closed

The Unit CRS has determined that conditions requiring a Manual Trip have been met. Upon actuation of **both** RTGB Manual Trip Pushbuttons by the OAC, the reactor has failed to trip.

- No AUTOMATIC trip setpoints have been reached

Based on the above conditions, the "B" Bypass Breaker ____(1)____ could **NOT** contribute to the failure of the reactor to TRIP. The crew ____(2)____ required to enter EOP-E-0, "Reactor Trip or Safety Injection".

	(1)	(2)
A.	Shunt Trip Coil	is
B.	Under-voltage Coil	is
C.	Shunt Trip Coil	is NOT
D.	Under-voltage Coil	is NOT

<u>**Distractor Analysis**</u>

A. <u>Correct</u>. Given information provides applicant success path.

B. Incorrect. Answer is plausible due to being partially correct. The Bypass Breaker Undervoltage Coil is directly affected during actuation of Manual Trip Pushbuttons and its failure could be a contributing cause for the failure of the reactor to trip. (Ref ST-011, Page 16)

C. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “B” and “D” distractor analyses.

D. Incorrect. Answer is plausible due to being partially correct. EOP-E-0 entry condition 2.e. is a plant specific requirement that mandates procedural entry in the event of an operator demanded reactor trip (i.e. actuation of manual trip pushbuttons). (Ref EOP-E-0-BD, Page 5)

References

- ST-011, Revision 4, Page 16
- EOP-E-0, Revision 5, Page 5
- EOP-E-0-BD, Revision 5, Page 5

KA Match

Applicant is given a situation where a diagnosis of plant conditions and systems knowledge is required to determine RPS bypass breaker impact and use of controlling procedure during a failure of the plant to trip.

Cognitive Level

High

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 39

012A4.07

Reactor Protection
Ability to manually operate and/or monitor in the control room: M/G set breakers
RO 3.9/SRO 3.9
CFR 41.7

Initial Conditions:

- Power is 100% RTP
- Annunciator APP-009-B8, “480V NORM BUS OVLD” is received
- Annunciator APP-009-E7, “480V GRD FAULT” is received

Current Conditions:

- Upon investigation, the Field AO reports that the following breakers have TRIPPED and are OPEN:
  - 52/15B, “MAIN BKR. 480V BUS 3”
  - 52/14A, “ ‘B’ ROD DRIVE MG”
- In addition, the Field AO reports that the 480V Bus 3 “GROUND FAULT RELAY FLAG” has actuated.

Based on the above conditions, receipt of annunciator APP-005-E6, “ROD CONT MG SETS TRIPPED” ____(1)____ expected. Upon the tripping of breaker 52/15B the ground fault condition cleared. With no further operator action APP-009-E7 will ____(2)____ extinguish.

	(1)	(2)
A.	is	Automatically
B.	is	NOT
C.	is NOT	Automatically
D.	is NOT	NOT

Distractor Analysis

A. Incorrect. Answer is plausible due to being partially correct. Annunciator APP-009-E7 will not extinguish until the ground fault relay is reset for faults on 480V Bus 3. This answer would be correct if the affected bus were 480V Bus 4. (Ref APP-009, Page 48)

B. Correct. Given information provides applicant success path.

C. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “A” and “D” distractor analyses.

D. Incorrect. Answer is plausible due to being partially correct. Opening **either** MG Set supply breaker on overload (given condition) will result in receipt of this annunciator. (Ref APP-005, Page 39)

References

- APP-009, Revision 56, Pages 23 and 48
- APP-005, Revision 39, Page 39

KA Match

Applicant is given a situation where a diagnosis of annunciators and systems knowledge is required to determine plant response.

Cognitive Level

Low

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 40

013K2.01

013 K2.01
Engineered Safety Features Actuation System (ESFAS)
Knowledge of bus power supplies to: ESFAS/safeguards equipment control
3.6*/3.8   (41.7)

Given the following conditions

- The unit is operating at 100% power.
- PZR Pressure channel PT-455 is failed, with all bistables in the TRIPPED condition.
- A subsequent electrical fault results in loss of Instrument Bus 3.

Which ONE (1) of the following describes the impact that the loss of Instrument Bus 3 has on the plant? A reactor trip occurs, __________________.

A. but **NO** SI occurs

B. and an SI occurs, but **ONLY** Train 'A' Engineered Safeguards loads are automatically started by the sequencers

C. and an SI occurs, but **ONLY** Train 'B' Engineered Safeguards loads are automatically started by the sequencers

D. and an SI occurs, with **BOTH** trains of Engineered Safeguards loads automatically started by the sequencers

<u>Distractor Analysis</u>

A. <u>Incorrect</u>. An SI occurs because 2 of 3 Pressurizer Pressure channels are tripped/deenergized, making up the <1715# SI logic. Plausible if the applicant remembers the 1844# reactor trip but forgets that the SI also comes from these instruments.

B. **<u>CORRECT</u>**. An SI occurs on 2 of 3 pressure channels, but with loss of IB3, only the ‘A’ Train sequencer can start its loads. Power is lost to Train ‘B’ interposing relays, which are fed from IB3.

C. <u>Incorrect</u>. An SI occurs, but with loss of IB3 the interposing relays cannot start their ESF loads.

D. <u>Incorrect</u>. An SI occurs, but ‘B’ Train interposing relays have no power, so BOTH trains actuating is incorrect.

References:
ST-006, Engineered Safety Features System, Rev. 11
ST-011, Reactor Protection System, Rev. 4

KA Match:
The KA is matched because the applicant must recall the power supply to the ‘B’ Train of ESFAS interposing relays and determine the impact of its loss.

Cognitive Level: **High**

Source of Question: Robinson 2001 NRC exam question 81 (SRO question). Changed conditions so ESFAS Train B became the correct answer; changed the order of first and last distractors.

QUESTION 41

022A3.01

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	A3.01	
	Importance Rating	4.1	

Proposed Question:

Unit 2 plant conditions:

0400

- 100% power.
- Containment Air Recirculation Cooling units HVH-1, HVH-3, and HVH-4 are in operation.
- Containment Air Recirculation Cooling unit HVH-2 has been stopped for breaker maintenance that will be performed on the next shift.

0500

- A large break LOCA has occurred
- SI has actuated

0510

- The SI signal has been reset

At 0515, the normal air inlet damper for HVH-1 is __(1)__ and the status of HVH-2 fan is __(2)____.

Which one of the following completes the statements above?

A	(1) closed (2) off
B	(1) closed (2) on
C	(1) open (2) off
D	(1) open (2) on

Proposed Answer: **B**

K/A Match Analysis

022 Containment Cooling System (CCS), A3.01

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

Requires knowledge of the automatic initiation of safeguards mode of the containment cooling system.

Answer Choice Analysis

A. (1) CORRECT, The normal air inlet dampers close on a SI SIGNAL and relays will prevent them from automatically reopening when the SI SIGNAL is reset.
(2) INCORRECT, plausible because the four containment cooling units are single speed fans and normally all in operation therefore no actuation signal is required.

B. (1) CORRECT
(2) CORRECT, any containment cooling units that are not in operation will receive a start signal from the SI circuit.

C. (1) INCORRECT, plausible because a plant modification has been performed which installed relays to prevent the reopening of the dampers upon SI reset. Prior to the modification the dampers would have opened; also the switch to reset the dampers for normal operation is located on the DC relay racks in the computer room. There also is a damper at the fan discharge which operates in conjunction with the fan.
(2) INCORRECT

D. (1) INCORRECT
(2) CORRECT

Technical Reference(s):

Containment HVAC, SD-037, Revision 9
ENGINEERED SAFETY FEATURES SYSTEM, SD-006, Revision 11
SAFETY INJECTION SYSTEM, SD-002, Revision 15

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: Last NRC Exam: NEW

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis _X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 42

026G2.1.31

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	G2.1.31	
Importance Rating	4.6	

Proposed Question:

The unit tripped from 100% power with the following conditions:

- Safety Injection is in progress due to a Large Break LOCA.
- Containment pressure peaked at 15 psig and currently is 6 psig and decreasing slowly.
- The following annunciators are in alarm:
  - APP-02 F2 SPRAY ADD TANK LO LEVEL
  - APP-02 A3 RWST HI/LO LVL
- Spray additive tank level is 10% and decreasing slowly

The operators will close CV spray additive tank discharge valves, SI-845A and SI-845B ___(1)____ and go to EPP-9 TRANSFER TO COLD LEG RECIRCULATION ___(2)___.

Which one of the following completes the statement above?

A	(1) at this time (2) when alarm APP-02 B3 RWST LO-LO LVL actuates
B	(1) at this time (2) at this time
C	(1) when containment pressure is less than 4 psig (2) when alarm APP-02 B3 RWST LO-LO LVL actuates
D	(1) when containment pressure is less than 4 psig (2) at this time

Proposed Answer: **C**

K/A Match Analysis

026 Containment Spray System (CSS), G2.1.31

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Requires the ability to determine the expected indications on control room instruments and determine the desired plant lineup based on those indications.

Answer Choice Analysis

I. INCORRECT. Plausible because the CV spray tank additive valves get an auto open signal but no auto close signal, the applicant may assume the need to isolate the additive tank upon receipt of the low level alarm. Second part is plausible because applicant may assume the A3 RWST HI/LO LVL alarm would actuate on a Tech Spec low level and the RWST LO-LO LVL is the level at which EPP-9 would be started.

J. INCORRECT. Second part is correct.

K. CORRECT. In EOP E-1, the additive tank discharge valves would be closed when containment pressure is less than 4 psig as part of stopping containment spray. In EPP-9 they are closed only when tank level is 0%.

L. CORRECT.

Technical Reference(s):
EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, REVISION 2
APP-002 rev 66

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: New

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 43

039A4.04

K/A number and description:

039A4.04	039 Main and Reheat Steam System (MRSS) A4: Ability to manually operate and/or monitor in the control room: A4.04: Emergency Feedwater pump turbines

Question:

Given the following plant conditions:

- A loss of offsite power and a reactor trip has occurred
- APP-007-F5, SD AFW PMP LO DISCH PRESS TRIP alarms and remains locked in
- Two minutes later the pump has been verified to be coasting down locally due to an overspeed trip

Which one of the following completes the statements below?

SDAFW Pump Steam Shutoff Valves, V1-8A, V1-8B, and V1-8C should be ___(1)___ at this time. The BOP operator is required to ___(2)___ closure of SDAFW Pump Discharge Valves, V2-14A, V2-14B, and V2-14C.

A. (1) open
(2) verify automatic

B. (1) open
(2) perform manual

C. (1) closed
(2) verify automatic

D. (1) closed
(2) perform manual

Proposed Answer: D

Distractor Analysis:

A. Incorrect. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. Plausible because on the overspeed trip the SDAFW Pump throttle valve closes to isolate the steam supply, so it is not necessary for the steam supply valves to close to stop the pump. The discharge valves do not close automatically, but APP-007-F5 directs manual closure. Plausible that an auto signal would close the valve to isolate a discharge line break that caused the low discharge pressure trip, since some valves auto close on the trip.

B. Incorrect. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. Plausible because on the overspeed trip the SDAFW Pump throttle valve closes to isolate the steam supply, so it is not necessary for the steam supply valves to close to stop the pump.

C. Incorrect. The discharge valves do not close automatically, but APP-007-F5 directs manual closure. Plausible that an auto signal would close the valve to isolate a discharge line break that caused the low discharge pressure trip, since some valves auto close on the trip.

D. Correct. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. APP-007-F5, directs manual closure of V2-14A, B, and C.

Reference(s)

- ST-042
- APP-007, Rev 42. CONDENSATE & FEEDWATER, Panel F5

K/A Match discussion:

This question meets the K/A because the candidate is required to determine proper pump configuration in the control room for the conditions given.

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

Question Source:	Bank:	___X___	
	Modified Bank:	_______	(Note changes or attach parent)
	New:	_______	

Comment: Bank source was RNS NRC exam 2013-301, Q 42

SRO Only Basis: N/A

QUESTION 44
059A2.12

Main Feedwater
Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of feedwater regulating valves
RO 3.1/SRO 3.4
CFR 41.5/43.5

Initial Conditions:

- Reactor power is at 62% RTP
- ‘C’ S/G water level is stable at 52%
- FCV-498, “S/G ‘C’ FWRV” has been placed in local-handwheel control in accordance with OP-403, “FEEDWATER SYSTEM” for level control circuit maintenance
- Field personnel have reached OP-403 step 8.4.3.2.n, “INITIATE the required maintenance activities.”

Current Conditions:

- I&C Maintenance reports that the FCV-498 Bailey Positioner has failed and is out-putting a max air signal

Based on the above conditions, FCV-498 ____(1)____ reposition.

Should “MFP A” trip during the above conditions, AOP-010, “MAIN FEEDWATER/CONDENSATE MALFUNCTION” will direct Operators to ____(2)____ .

	(1)	(2)
A.	will	perform a Reactor Trip
B.	will	stabilize reactor power < 60%
C.	will NOT	perform a Reactor Trip
D.	will NOT	stabilize reactor power < 60%

Distractor Analysis

A. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “B” and “C” distractor analyses.

B. Incorrect. Answer is plausible due to being partially correct. The instrument air signal to FCV-498 is isolated (via IA-3122) when step 8.4.3.2.n is reached, therefore the Bailey Positioner failure has no effect on FCV-498. (Ref ST-027, Page 10 and OP-403, Pages 32 & 33)

C. Incorrect. Answer is plausible due to being partially correct. A failure of a MFP with the given conditions (i.e. Reactor Power less than 70%) does not satisfy the reactor trip criteria of AOP-010. (Ref AOP-010, Pages 3 - 5)

D. Correct. Given information provides applicant success path.

References

- ST-027, Revision 4, Page 10
- OP-403, Revision 49, Pages 32 & 33
- AOP-010, Revision 31, Pages 3 - 5

KA Match

Applicant is given a situation where a diagnosis of given conditions and systems knowledge is required to determine plant response and crew mitigation strategy.

Cognitive Level

High

Source of Question

Modified, 2013-301

SRO Only Basis

Not applicable

QUESTION 45
061K6.02

Question 45
061K6.02
Auxiliary/Emergency Feedwater
Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps
RO 2.6/SRO 2.7
CFR 41.7

Current Conditions:

- The CST has developed a leak
- The reactor is in Mode 1; Mode 3 preparations are currently in progress

OP-402, "AUXILIARY FEEDWATER SYSTEM" states that once CST level is below ____(1)____ %, the ____(2)____ Driven AFW pump(s) require(s) venting prior to use, if not already in operation.

	(1)	(2)
A.	10	Motor
B.	10	Steam
C.	34	Motor
D.	34	Steam

Distractor Analysis

A. Incorrect. Plausible if applicant misapplies incorrect answers to question, see "B" and "C" distractor analyses.

B. Incorrect. Answer is plausible due to being partially correct. Per OP-402, actions required at 10% CST level refer to placing a backup water supply in service during AFW operation. (Ref OP-402, Page 8)

C. Incorrect. Answer is plausible due to being partially correct. Per OP-402, the Steam Driven AFW pump requires venting due to the possibility of draining the suction line and potential air binding of the pump if operated. (Ref OP-402, Page 25)

D. Correct. Given information provides applicant success path.

References

- OP-402, Revision 86, Pages 8 & 25

KA Match

Applicant is given a situation where equipment operating knowledge is required to determine crew mitigation strategy to off-normal conditions.

Cognitive Level

Low

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 46
062A1.01

062 A1.01 A.C. Electrical Distribution
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: significance of D/G load limits. 3.4/3.8 (41.5, 45.5)

Emergency Diesel Generator (EDG) amperage shall **NOT** exceed __(1)__, to prevent exceeding the current rating of the __(2)__.

	(1)	(2)
A.	3200 amps	EDG output breaker
B.	3200 amps	generator stator windings
C.	4000 amps	EDG output breaker
D.	4000 amps	generator stator windings

Correct answer: C

Distractor Analysis

3200 amps is plausible because it is the max current for the EDG's normal load limit of 2500 kW. Multiple places in OP-604, Diesel Generators "A" and "B", direct you to not exceed 3200 amps when approaching 2500 kW. This figure is also given in Attachment 10.8, Diverse Load Indications.

4000 amps is the correct answer, supported by OP-604 P&L 5.7: "4,000 amps on the Generator shall **NOT** be exceeded to ensure the 4000 amp rating on the EDG Output breaker is **NOT** exceeded."

To protect the generator stator windings from overcurrent is plausible because this is a pretty standard protection for generators. There is a generator overcurrent relay mounted on each Generator Control Panel, lending further credence to this distractor.

References:
OP-604, Diesel Generators "A" and "B", Rev. 99
SD-005, Emergency Diesel Generators, Rev. 17

KA Match:
The KA is matched because the applicant has to demonstrate knowledge of a D/G load limit and its potential impact on the AC distribution system.

Cognitive Level: **Low**

Source of Question: **New**

QUESTION 47
062K1.04

062 K1.04 A.C. Electrical Distribution
Knowledge of the physical connections and/or cause-effect relationships between the ac distribution system and: off-site power sources. 3.7/4.3 (41.2 to 41.9)

- The plant is at 100% power, steady-state.
- A fault in the 230kV switchyard sends trip-open signals to generator output breakers 52/8 & 52/9.
- Breaker 52/8 fails to open.

1) Which generator lockout relay(s) will actuate **directly from 52/8 failure to open** (do not infer later signals that may actuate either or both lockout relays).

2) If the lockout relay(s) is/are reset with the 52/8 failure-to-open signal still present, __**(2)**__.

A. (1) 86P **AND** 86BU will actuate.
(2) one or more 4kV bus supply or tie breakers will reposition or attempt to.

B. (1) 86P **AND** 86BU will actuate.
(2) there will be no effect on 4kV bus alignment.

C. (1) **Only** 86P will actuate.
(2) one or more 4kV bus supply or tie breakers will reposition or attempt to.

D. (1) **Only** 86P will actuate.
(2) there will be no effect on 4kV bus alignment.

Distractor Analysis

Question tests lessons-learned from the 2010 RNP 2nd electrical fire.

First part is general knowledge: 3 causes are common to 86P & BU (52/8 or 9 not opening, and Turbine trip), 86P has 6 other causes/trips, and 86BU has 3. Even though it might be easily remembered that 52/8 & 9 are common to both, I chose that to better hit the “off-site power sources” part of the KA.

Second part tests knowledge of effects of resetting 86P/BU. During the fire event, a fault on 4kV Bus 5 caused 52/19, the Bus 4 feed from Bus 3 to open because Bus 5’s supply breaker 52/24 did not open per design. Resetting the lockout relays with a UAT fault pressure locked in caused 52/19 to re-close, as it would for a fast-bus transfer. In this question though, all 4kV breakers remain in their normal post-trip alignment so none should try to reposition.

First part: As discussed above, 52/8 or 9 failure to open is an actuation signal for **both** lockout relays. “Only 86P” is therefore incorrect.

Second part: There will be no effect on 4kV bus alignment. The 86/P & BU lockout relay handswitches in the control room can be taken to Reset, but they will not lock in place. This will momentarily reset the relays however, and when they trip back open they’ll send a fast dead bus transfer signal, but the affected breakers are already in their required positions (52/17 & 20 open, 52/12 & 19 closed). So no realignment (or attempted realignment) will occur.

References:
SD-034, MAIN GENERATOR AND AUXILIARIES SYSTEM, Rev. 12
SD-039, 230/4 KVAC ELECTRICAL SYSTEM, Rev. 18

KA Match:
The KA is matched because knowledge of the physical connections to off-site power sources is required (4kV breakers 52/7, 12, 19, & 2), as well as a cause-effect relationship between the ac distribution system and off-site power sources is required (230kV switchyard breaker protection scheme leading to trip of main generator).

Cognitive Level: **High**
First part of the question is pure memory, but the second part is higher cognitive level as the applicant must predict an impact based on analysis of given conditions.

Source of Question: **New**

QUESTION 48
063A1.01

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	A1.01	
Importance Rating	2.5	

Proposed Question:

Unit 2 plant conditions:

- A Loss of Offsite Power has occurred.
- BOTH EDGs have failed to auto start and cannot be manually started.
- EPP-1, Loss of All AC Power, has been implemented.

In accordance with EPP-1, low priority loads __(1)___ required to be shed from Instrument Buses 2 and 3 to minimize the discharge rate of station batteries.

Assuming battery load remains constant, battery current will __(2)__ as terminal voltage lowers.

A	(1) are (2) rise
B	(1) are (2) lower
C	(1) are NOT (2) rise
D	(1) are NOT (2) lower

Proposed Answer: **A**

K/A Match Analysis

063 D.C. Electrical Distribution, A1.01

Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate.

Requires knowledge of the effect of lowering battery voltage on current (discharge rate).

Answer Choice Analysis

M. (1) CORRECT, EPP-1 attachment 2 directs the shedding of loads from DC buses to conserve DC battery power, including instrument buses 2 and 3.
(2) CORRECT, Current will rise, P=VI, in this case Power (P) will remain constant and Voltage (V) will lower as the battery discharges therefore the Current (I) must rise to maintain the same Power.

N. (1) CORRECT
(2) INCORRECT, if the candidate misuses the equation P=VI. If the candidate uses V=IP, which is similar to V=IR, they would say that current has to lower. An applicant also could assume incorrectly that as battery capacity is used, current would have to lower.

O. (1) INCORRECT. Plausible because EPP-1, Attachment 2, contains CAUTION that states the following: Inverter A AND Inverter B should NOT be shed to ensure power is available to Instrument Buses 2 and 3.
(2) CORRECT

P. (1) INCORRECT
(2) INCORRECT

Technical Reference(s):

EPP-1, LOSS OF ALL AC POWER, Revision 51 and Attachment 2
EPP-1-BD, EPP-1 BASIS DOCUMENT, Revision 51A
ST-038, DC-ELECTRICAL SYSTEM, Revision 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank Robinson 063 A1.01 001 (parent attached)

Question History: Last NRC Exam: 2011-302 Question 50, half of the question was used.

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 49
064A3.13

K/A number and description:

064A3.1.3

064	Emergency Diesel Generator
A3.13	Rpm controller/megawatt load control (breaker-open/breaker-closed effects)

Question:

DG "A" auto started and loaded due to a Loss of Offsite Power. The DG speed control ____(1)_____ aligned with 0% percent droop.

Following the Blackout Sequencer loading, the 480 Emergency Bus E-1 frequency indicates 58.9 Hz. With no additional action, the speed of DG "A" ___(2)___be adjusted from the Generator Control Panel using the EDG "A" Speed Control Switch in accordance with OP-604.

	(2)	(2)
E.	is	can
F.	is	can NOT
G.	is NOT	can
H.	is NOT	can NOT

Distractor Analysis:

Proposed Answer: A

A. Correct. The first part is correct; the normal standby alignment has the Woodward governor droop setting at 0%; as a result the generator will behave as an isochronous machine and speed will remain constant with varying load. The second part is correct; OP-604, Attachment 10.10 is available to adjust speed if required following an auto start.

B. Incorrect. The first part is correct; the normal standby alignment has the Woodward governor droop setting at 0%; as a result the generator will behave as an isochronous machine and speed will remain constant with varying load. The second part is incorrect; OP-604, Attachment 10.10 is available to adjust speed if required following an auto start. The answer is plausible because the applicant may assume with the auto start that the ability to manually control speed from the Generator Control Panel has been blocked without additional operator action.

C. Incorrect. The first part is incorrect; the normal standby alignment has the Woodward governor droop setting at 0%; as a result the generator will behave as an isochroous machine and speed will remain constant with varying load on an emergency start. The answer is plausible because the EDG will be operated with a 50% droop when run in parallel with the normal power supply for testing, etc. The second part is correct; OP-604, Attachment 10.10 is available to adjust speed if required following an auto start.

D. Incorrect. The first part is incorrect; the normal standby alignment has the Woodward governor droop setting at 0%; as a result the generator will behave as an isochroous machine and speed will remain constant with varying load on an emergency start. The answer is plausible because the EDG will be operated with a 50% droop when run in parallel with the normal power supply for testing, etc. The second part is incorrect; OP-604, Attachment 10.10 is available to adjust speed if required following an auto start. The answer is plausible because the applicant may assume with the auto start that the ability to manually control speed from the Generator Control Panel has been blocked without additional operator action.

Reference(s)

- Operations Procedure, OP-604, Diesel Generators “A” and “B”, rev 99.
- Student Training, ST-005.

K/A Match discussion:

The K/A is matched because the question involves knowledge of EDG speed control circuit operation during automatic and manual operation.

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

Question Source:	Bank:	______
	Modified Bank:	_______ (Note changes or attach parent)

New: ___X__

SRO Only Basis: N/A

QUESTION 50
064G2.4.35

K/A number and description:

064G2.4.35

064	Emergency Diesel Generator
G2.4.35	Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects

Question:

The unit has entered EPP-1, Loss of All AC Power.

- EDG “A” is unavailable due to maintenance
- EDG “B” failed to start.
- The Inside AO was dispatched to locally start the “B” EDG.
- EDG “B” LOCAL-REMOTE switch was placed in the LOCAL position at the DG Engine Control Panel

When the AO depresses the Local Engine Start pushbutton, the engine _____(1)_____ automatically prelube.

Assuming the EDG “B” is successfully started and the LOCAL-REMOTE switch remains in the LOCAL position, the EDG “B” output breaker 52/27B _____(2)_____ be closed from the RTGB.

	(3)	(2)
I.	will	can
J.	will	can NOT
K.	will NOT	can
L.	will NOT	can NOT

Distractor Analysis:

Proposed Answer: C

A. Incorrect. The first part is incorrect. Plausible because when the diesel is started from the RTGB, the prelube pump starts and is controlled by a 4 minute 30 second TD relay; after which the EDG will start. The second part is correct; EDG output breaker operation remains available from the RTGB even with the L-R switch in LOCAL.

B. Incorrect. The first part is incorrect. Plausible because when the diesel is started from the RTGB, the prelube pump starts and is controlled by a 4 minute 30 second TD relay; after which the EDG will start. The second part is incorrect EGD output breaker operation remains available from the RTGB even with the L-R switch in LOCAL. The answer is plausible because the applicant may assume breaker control along with EDG start control is transferred local.

C. CORRECT. The first part is correct; when the engine is started from any other location from the RTGB (or automatically), it will not be automatically prelubed. The second part is correct; EDG output breaker operation remains available from the RTGB even with the L-R switch in LOCAL.

D. Incorrect. The first part is correct; when the engine is started from any other location from the RTGB (or automatically), it will not be automatically prelubed. The second part is incorrect EGD output breaker operation remains available from the RTGB even with the L-R switch in LOCAL. The answer is plausible because the applicant may assume breaker control along with EDG start control is transferred local.

Reference(s)

- Student Training, ST-005 (based on System Description SD-005).
- End Path Procedure, EPP-1, Loss of All AC Power, Rev 55

K/A Match discussion:

The K/A is matched because the question involves knowledge of EDG local operation during an emergency.

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

Question Source: Bank: _______
Modified Bank: _______ (Note changes or attach parent)
New: ___X___

SRO Only Basis: N/A

QUESTION 51
073K4.01

073 K4.01 [Originally was K4.02, but not applicable at RNP.]
Process Radiation Monitoring System
Knowledge of PRM system design feature(s) and/or interlock(s) which provide for:
Release termination when radiation exceeds setpoint. 4.0/4.3 (41.7)

Given the following:

- The plant is operating at 100% RTP.
- A release of Waste Condensate Tank (WCT) "A" is in progress.
- APP-036-E7, RAD MONITOR TROUBLE, is received.
- BOP reports the FAIL light for R-18, LIQUID WASTE DISPOSAL EFFLUENT monitor, is ON.

Which ONE (1) of the following describes the status of RCV-018, LIQUID WASTE RELEASE ISOLATION VALVE?

RCV-018 will...

A. NOT automatically close. The release must be terminated manually.

B. automatically close when the monitor FAIL light is illuminated.

C. NOT automatically close, and CANNOT be closed from the Waste Disposal Panel.

D. automatically close, and must be reset by cycling the valve controller's potentiometer.

<u>Distractor Analysis</u>

A: **Correct** - Fail light means loss of power and/or loss of indication. Valve will NOT close. Release must be manually terminated.

B: Incorrect - RCV-018 will close on a High Radiation Alarm, NOT a fail light.

C: Incorrect - RCV-018 can be operated at any time with control switch on the Waste Disposal Panel.

D: Incorrect - RCV-018 will NOT Automatically CLOSE and is controlled by the CLOSE-AUTO-OPEN switch, not a potentiometer. Plausibility lent by RCV-014 being controlled by a potentiometer.

References:
ST-019, Radiation Monitoring System, Rev. 0
APP-036-E7, Rev. 85

KA Match:
The KA is matched because applicant must demonstrate knowledge of PRM system design feature and interlock which provides for release termination when radiation exceeds setpoint.

Cognitive Level: **Low**

Source of Question: Robinson 2008 NRC exam

QUESTION 52
076K2.04

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	076K2.04
	Importance Rating	2.5

076K2.04 – Knowledge of the bus power supplies to reactor building closed cooling water

Proposed Question:

Plant conditions are as follows:

- Operating at 100% power
- 'B' EDG is unavailable due to emergent maintenance
- All other systems are in their normal at-power configuration

The "SU TRANSF OVLD/ PHASE▲ TRIP" alarm energizes due to an overload of breaker 52/17.

Thirty seconds after the above alarm is received, which of the following CCW pumps can be manually started from the main control room?

a. 'A' CCW pump **only**

b. 'B' CCW pump **only**

c. 'C' CCW pump **only**

d. 'A' **and** 'B' CCW pumps

Proposed Answer:

b. 'B' CCW pump **only**

Explanation:

This question requires the applicant to evaluate plant conditions with the presence of an alarm that indicates a loss of the startup transformer (SUT) and, based on evaluation, determine the effects on the availability of power to the CCW pumps.

a. **INCORRECT** – The 'A' CCW pump is normally supplied its power from the DS Bus which receives its power from SST 2C (from 4160VAC Bus 3 and the SUT). Thus, on the loss of the SUT, the DS Bus will de-energize until it is re-energized by the DSDG. The DSDG will not re-energize the DS Bus within the 30 second time frame presented in the question stem. Plausible if applicant does not factor in the DSDG time delayed start on the loss of voltage to the DS Bus.
b. **CORRECT** – The 'B' CCW pump receives its power from the E-1 bus via SST 2F and Bus 2 which is normally supplied from the Unit Auxiliary Transformer (UAT) at power. The loss of the SUT will not have an effect on the power to the E-1 bus.
c. **INCORRECT** – At power, the 'C' CCW pump normally receives its power from E-2 via SST 2G and 4160 VAC Bus 3 via the SUT. When the SUT is lost, Bus 3 loses power and would normally be re-energized by the 'B' EDG after it completes its start sequence. In this case, the 'B' EDG is unavailable, thus the 'C' CCW pump will not be available for manual start. Plausible if the applicant does not recall that the 'C' CCW pump is powered from the E-2 bus.
d. **INCORRECT** – The 'B' CCW pump receives its power from the E-1 bus via SST 2F and Bus 2 which is normally supplied from the Unit Auxiliary Transformer (UAT) at power. The loss of the SUT will not have an effect on the power to the E-1 bus. The 'A' CCW pump is normally supplied its power from the DS Bus which receives its power from SST 2C (from 4160VAC Bus 3 and the SUT). Thus, on the loss of the SUT, the DS Bus will de-energize until it is re-energized by the DSDG. The DSDG will not re-energize the DS Bus within the 30 second time frame presented in the question stem. Plausible if applicant does not factor in the DSDG time delayed start on the loss of voltage to the DS Bus.

Technical Reference(s): APP-009, Rev. 56
ST-013, "Component Cooling Water System"
ST-039, Revision 4a, "230/4KVAC Electrical System"
ST-056, Revision 3, "Dedicated S/D System…"

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: comprehension or analysis

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant needs to evaluate a set of given plant conditions and, based on evaluation, determine the availability of CCW pumps based on the effects of losing their associated power supplies.

QUESTION 53
078K1.05

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	078K1.05
	Importance Rating	3.4

078K1.05 – Knowledge of the physical connections and/or cause effect relationships between the IAS and MSIV air

Proposed Question:

Which one of the following completes the statement below regarding the design of the MSIV accumulators as stated in ST-025, Main Steam System?

MSIV accumulators are designed to maintain a minimum of ____(1)____ psig for at least ____(2)____ minutes.

	(1)	(2)
a.	58	15
b.	58	30
c.	85	15
d.	85	30

Proposed Answer:

b. 58 30

Explanation:

This question requires the applicant to demonstrate knowledge of the interrelationship between the instrument air system and the main steam isolation valves. (specifically, the design basis for the size of the MSIV accumulators)

a. **<u>INCORRECT</u>** – Per ST-025, the MSIV accumulators are required to maintain 58 psig of air pressure for 30 minutes with the instrument air header isolated. Plausible if the applicant confuses the design basis time requirement for the accumulators for the MSIVs with that of the EHC accumulators.
b. **<u>CORRECT</u>** – Per ST-025, the MSIV accumulators are required to maintain 58 psig of air pressure for 30 minutes with the instrument air header isolated.
c. **<u>INCORRECT</u>** – Per ST-025, the MSIV accumulators are required to maintain 58 psig of air pressure for 30 minutes with the instrument air header isolated. Plausible if the applicant confuses the design basis time requirement for the accumulators for the MSIVs with that of the EHC accumulators OR confuses the pressure requirement with the entry criteria for AOP-017 “Loss of Instrument Air.”
d. **<u>INCORRECT</u>** – Per ST-025, the MSIV accumulators are required to maintain 58 psig of air pressure for 30 minutes with the instrument air header isolated. Plausible if the applicant confuses the pressure requirement with the entry criteria for AOP-017 “Loss of Instrument Air.”

Technical Reference(s): ST-025, “Main Steam System” Revision 4
UFSAR, Revision 15
AOP-017, “Loss of Instrument Air” Revision 40

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: fundamental knowledge or memory

10 CFR Part 55 Content: 55.41 (4)

K/A match: The K/A is matched since the applicant is required to demonstrate knowledge of the interrelationship between the instrument air system and the main steam isolation valves (design basis for MSIV accumulator size).

QUESTION 54
078K3.02

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	078K3.02
	Importance Rating	3.4

078K3.02 – Knowledge of the effect that a loss or malfunction of the IAS will have on systems having pneumatic valves and controls

Proposed Question:

Given the following:

- The plant was operating at 100% power
- A catastrophic failure of Instrument Air IA) header piping occurred
- IA header pressure is 5 psig and lowering
- APP-001-C1, “RCP THERM BAR COOL WTR HI FLOW,” is in alarm
- APP-001-D1, “RCP THERM BAR COOL WTR LO FLOW,” is in alarm
- Operators are concurrently executing AOP-017, “Loss of Instrument Air,” and EOP-E-0 “Reactor Trip or Safety Injection”
- ‘A’ Charging pump is running at minimum speed per AOP-017

Based on the above conditions:

- RCP seal leakoff temperature will ____(1)____.

Consider separately:

- With the instrument air header depressurized, in order to raise RCP seal injection flow per Section B of AOP-017, you shall ____(2)____.

	(1)	(2)
a.	be unaffected	throttle closed charging flow valve (HCV-121) from the Main Control Room
b.	be unaffected	locally throttle RCP seal water flow control valves
c.	rise	throttle closed charging flow valve (HCV-121) from the Main Control Room
d.	rise	locally throttle RCP seal water flow control valves

Proposed Answer:

d. rise locally throttle RCP seal water flow control valves

Explanation:

This question requires the applicant to assess plant conditions and determine the effects of an instrument air header rupture on the RCP seal leakoff temperature (with FCV-626 closed) as well as the means to throttle seal injection flow with a loss of air to HCV-121.

a. **INCORRECT** – The stem of the question provides the applicant with information that should help in determining that HCV-121 is failed open (and cannot be throttled in the control room), FCV-626 is closed (as determined by the alarm given), and only one charging pump running at minimum speed. In this configuration, seal injection flow will be robbed from the RCPs (since min speed is 18 – 22 gpm and the charging flowpath to the RCS is still aligned). Since seal injection to the RCPs will lower, coupled with closure of FCV-626, RCP pump components will be subjected to higher temperatures as hot RCS fluid enters the pump cavity and flows to the seals, thus making leakoff temperatures rise. Since HCV-121 is an air operated control valve, the valve cannot be throttled from the main control room. Plausible if the applicant does not determine that seal injection flow will go down OR if they do not remember that HCV-121 cannot be throttled from the control room with no air pressure.
b. **INCORRECT** – The stem of the question provides the applicant with information that should help in determining that HCV-121 is failed open (and cannot be throttled in the control room), FCV-626 is closed (as determined by the alarm given), and only one charging pump running at minimum speed. In this configuration, seal injection flow will be robbed from the RCPs (since min speed is 18 – 22 gpm and the charging flowpath to the RCS is still aligned). Since seal injection to the RCPs will lower, coupled with closure of FCV-626, RCP pump components will be subjected to higher temperatures as hot RCS fluid enters the pump cavity and flows to the seals, thus making leakoff temperatures rise. Section B of AOP-017, directs operators to locally throttle seal flow control flows for the RCPs if HCV-121 has failed open. Plausible if the applicant does not determine that seal injection flow will go down.
c. **INCORRECT** – The stem of the question provides the applicant with information that should help in determining that HCV-121 is failed open (and cannot be throttled in the control room), FCV-626 is closed (as determined by the alarm given), and only one charging pump running at minimum speed. In this configuration, seal injection flow will be robbed from the RCPs (since min speed is 18 – 22 gpm and the charging flowpath to the RCS is still aligned). Since seal injection to the RCPs will lower, coupled with closure of FCV-626, RCP pump components will be subjected to higher temperatures as hot RCS fluid enters the pump cavity and flows to the seals, thus making leakoff temperatures rise. Since HCV-121 is an air operated control valve, the valve cannot be throttled from the main control room. Plausible if the applicant does not remember that HCV-121 cannot be throttled from the control room with no air pressure.
d. **CORRECT** – The stem of the question provides the applicant with information that should help in determining that HCV-121 is failed open (and cannot be throttled in the control room), FCV-626 is closed (as determined by the alarms given), and only one charging pump running at minimum speed. In this configuration, seal injection flow will be robbed from the RCPs (since min speed is 18 – 22 gpm and the charging flowpath to the RCS is

still aligned). Since seal injection to the RCPs will lower, coupled with closure of FCV-626, RCP pump components will be subjected to higher temperatures as hot RCS fluid enters the pump cavity and flows to the seals, thus making leakoff temperatures rise. Section B of AOP-017, directs operators to locally throttle seal flow control flows for the RCPs if HCV-121 has failed open.

Technical Reference(s): AOP-017, Revision 40, Loss of Instrument Air
APP-001-C1, Revision 57
APP-001-D1, Revision 57
ST-001, RCS Revision 4
ST-021, CVCS Revision 2

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: Comprehension or analysis

10 CFR Part 55 Content: 55.41 (4)

K/A match: The K/A is matched since the applicant is required to determine the effect of a loss of instrument air supply pressure on RCP seal leakoff temperature and control of RCP seal injection flow.

QUESTION 55
103K4.04

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	K4.04	
Importance Rating	2.5	

Proposed Question:

Which one of the following describes a correct configuration for the Containment Main Personnel Access Hatch doors and equalizing valves while the airlock is in use for entering/exiting Containment?

A	Outer door OPEN Outer door equalizing valve OPEN Inner door CLOSED Inner door equalizing valve CLOSED
B	Outer door OPEN Outer door equalizing valve OPEN Inner door CLOSED Inner door equalizing valve OPEN
C	Outer door CLOSED Outer door equalizing valve OPEN Inner door OPEN Inner door equalizing valve CLOSED
D	Outer door CLOSED Outer door equalizing valve CLOSED Inner door OPEN Inner door equalizing valve CLOSED

Proposed Answer: **A**

K/A Match Analysis
103 Containment System, K4.04
Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Personnel access hatch and emergency access hatch.

Requires knowledge of the interlocks and design features for the containment personnel access hatch.

Answer Choice Analysis

A. CORRECT. Only one door is open at a time AND when outer door is open, its equalizing valve is open. AND when the inner door is closed, its equalizing valve is closed.
B. INCORRECT. Plausible because only one door is open at a time BUT both equalizing valves are open at the same time.
C. INCORRECT. Plausible because only one door is open at a time BUT the door that is closed has its equalizing valve open AND the door that is open has its equalizing valve closed.
D. CORRECT. Only one door is open at a time BUT the door that is open has its equalizing valve closed.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 2011-301 Farley Q 59 (HLT-34 5/24/2011)

Question History: Last NRC Exam: 2011-301 Farley Q 59 (HLT-34 5/24/2011)

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

Supporting References

QUESTION 56
002K6.03

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	002K6.03
	Importance Rating	3.1

002K6.03 – Knowledge of the effect or a loss or malfunction of reactor vessel level indication

Proposed Question:

During alignment of RVLIS for operation per OP-307, “Inadequate Core Cooling Monitor,” personnel performing the valve lineup incorrectly left the Seal Table RVLIS Sensor Isolation Valve (RC-592) CLOSED.

Which of the following RVLIS ranges would provide an accurate indication of vessel level?

a. Dynamic Head

b. Full Range

c. Upper Range

d. None, no RVLIS level indication will be accurate

Proposed Answer:

c. Upper Range

Explanation:

This question requires the applicant to recognize the effect of a mispositioned valve on RVLIS indication validity.

a. **<u>INCORRECT</u>** – With RC-592 left in the shut position, the low pressure inputs to the full range level transmitters AND the high pressure inputs to the dynamic head D/P transmitters would be isolated. Plausible if the applicant does not recognize which transmitters are affected by the mispositioned valve.
b. **<u>INCORRECT</u>** – With RC-592 left in the shut position, the low pressure inputs to the full range level transmitters AND the high pressure inputs to the dynamic head D/P transmitters would be isolated. Plausible if the applicant does not recognize which transmitters are affected by the mispositioned valve.
c. **<u>CORRECT</u>** – With RC-592 left in the shut position, the low pressure inputs to the full range level transmitters AND the high pressure inputs to the dynamic head D/P transmitters would be isolated. The low and high pressure inputs to the upper range level transmitters are not affected by the mispositioned valve.
d. **<u>INCORRECT</u>** – With RC-592 left in the shut position, the low pressure inputs to the full range level transmitters AND the high pressure inputs to the dynamic head D/P transmitters would be isolated. Plausible if the applicant does not recognize which transmitters are affected by the mispositioned valve.

Technical Reference(s): HBR2-9067, “RVLIS Flow Diagram,” Revision 8
OP-307, “Inadequate Core Cooling Monitor” Revision 13
ST-051, “Inadequate Core Cooling Monitor System” Revision 2

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: fundamental knowledge or memory

10 CFR Part 55 Content: 55.41 (7)

K/A match: The K/A is matched since the applicant is required to determine the effects of a loss of inputs to level transmitters on RVLIS indication validity.

QUESTION 57
014K3.02

K/A number and description:

014K3.02

014 Rod Position Indication System (RPIS)
K3 Knowledge of the effect that a loss or malfunction of the RPIS will have on the following:
K3.02 Plant computer

<u>Question:</u>

The Unit is commencing a Reactor Start Up and is pulling control groups “A” and “B” rods. At 25 steps on the “B” bank an IRPI module for a “B” group rod fails low. Which one of the following describes the alarms and/or indications which would alert the operator to this condition?

Annunciator APP-005-F2 ROD BOTTOM, ROD DROP ___(1)___ be ILLUMINATED and an ERFIS printout for Rod Deviation ____(2)____ occur.

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

<u>Distractor Analysis:</u>

Proposed Answer: C

E. Incorrect. The first part is incorrect, a F2 annunciator will not be received ≤ 35 steps. The second part is correct, with “B” Bank < 200 steps and deviation > 7.5 inches, the Rod Deviation (ERFIS printout) will occur.

F. Incorrect. The first part is incorrect, a F2 annunciator will not be received ≤ 35 steps. The second part is incorrect, with “B” Bank < 200 steps and deviation > 7.5 inches, the Rod Deviation (ERFIS printout) will occur.

G. Correct. The first part is correct, a F2 annunciator will not be received ≤ 35 steps. The second part is correct, with “B” Bank < 200 steps and deviation > 7.5 inches, the Rod Deviation (ERFIS printout) will occur.

H. Incorrect. The first part is incorrect, a F2 annunciator will not be received ≤ 35 steps. The second part is incorrect, with “B” Bank < 200 steps and deviation > 7.5 inches, the Rod Deviation (ERFIS printout) will occur.

Reference(s)

- INDIVIDUAL ROD POSITION INDICATION SYSTEM, SD-009, Revision 2
- Annunciator Panel Procedure: APP-005, NIS and Reactor Control
- Abnormal Operating Procedure: AOP-001, Malfunction of Reactor Control System R28

K/A Match discussion:

The K/A is matched because the question involves conditions related to an RPIS failure and the associated printout on the plant computer as well as an associated annunciator.

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

Question Source: Bank: ______
Modified Bank: ________ (Note changes or attach parent)
New: ___X__

SRO Only Basis: N/A

QUESTION 58
017A2.02

Question 58
017A2.02
In-Core Temperature Monitor
Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Core damage
RO 3.6/SRO 4.1
CFR 41.5/43.5

A Loss of Coolant Accident (LOCA) has been in progress for the last two hours. You have been assigned to monitor plant parameters as indicated on the Main Control Room ICCM plasma displays.

Both Train 'A' and 'B' ICCM plasma displays contain the exact **same** indications.

- FULL RANGE RVLIS 100%
- SUBCOOL (T/C) 34^{o}F
- AVG 5 HIGH 605^{o}F
- SUBCOOL (RTD) 37^{o}F
- HIGH RTD 607^{o}F

Operators are monitoring the CSF-2, Core Cooling Critical Safety Function Status Tree and have reached the "RCS Subcooling" Decision Block.

In accordance with OMM-022, "EMERGENCY OPERATING PROCEDURES USER'S GUIDE" and based on the conditions above, CSFST parameter monitoring ____(1)____ be relaxed to 10-20 minutes.

When monitoring for the "RCS Subcooling" Decision Block, the ICCM plasma display value of ____(2)____ should be used per OMM-022.

	(1)	(2)
A.	may	SUBCOOL (T/C)
B.	may	SUBCOOL (RTD)
C.	may NOT	SUBCOOL (T/C)
D.	may NOT	SUBCOOL (RTD)

Distractor Analysis

A. Correct. Given information provides applicant success path.

B. Incorrect. Answer is plausible due to being partially correct. Per OMM-022, the correct subcooling indication is procedurally defined as the one supplied by thermocouples. (Ref OMM-022, Page 40)

C. Incorrect. Answer is plausible due to being partially correct. Per OMM-022, parameter monitoring relaxation can only be exercised when not involved in a Red or Orange CSF mitigation strategy. Misinterpretation of given conditions can lead applicant to any of the four CSF response categories. (Ref OMM-022, Page 23)

D. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “B” and “C” distractor analyses.

References
- OMM-022, Revision 43, Pages 23 and 40
- CSFST, Revision 6, Page 4
- ST-51, Revision 2, Page 17

KA Match
Applicant is given a situation where evaluation of plant parameters is required to determine performance of crew response strategy.

Cognitive Level
High

Source of Question
New

SRO Only Basis
Not applicable

QUESTION 59
027K2.01

027 K2.01
Containment Iodine Removal System
Knowledge of bus power supplies to: fans 3.1*/3.4* (41.7)

Which ONE (1) of the following is the power supply for HVE-3, CV Air Iodine Removal Fan?

A. MCC-5

B. MCC-6

C. MCC-9

D. MCC-10

Distractor Analysis

A. **Correct**. HVE-3 is powered from MCC-5 (Train A).

B. Incorrect. MCC-6 is ‘B’ Train vital power. Plausible because HVE-3 & 4 don’t have train designators, and there is no convention at RNP that odd numbers are ‘A’ Train, etc.

C. Incorrect. MCC-9 is vital power for Train B, fed from MCC-6. Many safety-related components are fed from these “secondary” MCCs, such as AFW valves, Service Water valves, etc.

D. Incorrect. MCC-10 is vital power from Train A, fed from MCC-5.

References:
EDP-003, MCC Buses, Rev. 56
ST-036, HVAC System, Rev. 3
ST-037, Containment HVAC, Rev. 2
Learning Objective CVHVAC005

KA Match:
The KA is matched because the applicant has to know what the power supply is to one of the two containment iodine removal fans.

Cognitive Level: **Low**

Source of Question: HBR 2009 NRC exam

Tier 2 / Group 2

QUESTION 60
029K4.02

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	K4.02	
	Importance Rating	2.9	

Proposed Question:

Unit 2 plant conditions:

- Containment particulate and gas monitors R-11 and R-12 are in service
- Containment Purge fan HVE-1A is running

If purge fan HVE-1A loses power ___(1)____.

If containment monitor R-11 alarms, manual operation of containment purge inlet and outlet valves ___(2)___be performed after resetting R-11.

Which one of the following completes the statements above?

A	(1) all containment purge inlet and outlet valves will automatically close (2) can
B	(1) all containment purge inlet and outlet valves will automatically close (2) can NOT
C	(1) containment purge fan HVE-1B will automatically start (2) can
D	(1) containment purge fan HVE-1B will automatically start (2) can NOT

Proposed Answer: **D**

K/A Match Analysis
029 Containment Purge System (CPS), K4.02
Knowledge of design feature(s) and/or interlock(s) which provide for the following: Negative pressure in containment.

Requires knowledge of interlocks associated with maintaining negative pressure in containment with the containment purge system.

Answer Choice Analysis

Q. (1) INCORRECT. Plausible because the valves will close on a Phase A or B signal and on R-11 or R-12 alarm.
(2) INCORRECT.

R. (1) INCORRECT
(2) CORRECT. If R-11 alarms, the R-11 alarm must be reset and then the Containment Ventilation Isolation Reset pushbutton must be pushed to allow valve and fan operation.

S. (1) CORRECT. Fans are interlocked so that if the running fan motor trips electrically, the other will start.
(2) INCORRECT. Plausible because the alarm on R-11 will need to be reset to allow operation.

T. (1) CORRECT
(2) CORRECT. Operation of the Containment Ventilation Isolation Reset pushbutton is required to allow operation of the containment purge system after the signal from R-11 is reset.

Technical Reference(s):

SD-037, Containment HVAC, Revision 9

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: New

Question Cognitive Level: Memory or Fundamental Knowledge _

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 _

Comments:

Supporting References

QUESTION 61
033G2.1.25

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	G2.1.1
	Importance Rating	3.8

G2.1.1 – Knowledge of conduct of operations requirements

Proposed Question:

The plant is operating at 100% power when the following events occur:

- Shift personnel in the Control Room Complex begin to cough and their eyes start to burn concurrent with identification of an unusual odor
- A Unit Operator reports that a large chemical spill has occurred in the Auxiliary Building and recommends that the building be evacuated due to the resultant eye and respiratory irritation being experienced in the area

Based on the above conditions, the Control Room operating crew should perform which of the following actions?

1. Immediately don SCBAs
2. Align the Control Room Complex ventilation system to Emergency Recirculation
3. Manually trip the reactor and main turbine
4. Report to the Old Fire Equipment Building

a. 1 **only**

b. 2 **only**

c. 1 **and** 2

d. 3 **and** 4

Proposed Answer:

c. 1 **and** 2

Explanation:

This question requires the applicant to assess plant conditions and determine the required actions for a toxic gas event in the control room complex that does not require control room evacuation per the requirements of AOP-004 “Control Room Inaccessibility”.

a. **INCORRECT** – The applicant should determine that the unusual odor in the Control Room Complex coupled with the report of a chemical spill in the Auxiliary Room does not meet the entry conditions for AOP-004. In this case the applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation. Only donning SCBAs does not meet the requirements of the procedure.
b. **INCORRECT** – The applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation. Only donning SCBAs does not meet the requirements of the procedure. Only re-aligning Control Room ventilation does not meet the requirements of the procedure.
c. **CORRECT** – The applicant should determine that the unusual odor in the Control Room Complex coupled with the report of a chemical spill in the Auxiliary Room does not meet the entry conditions for AOP-004. In this case the applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation.
d. **INCORRECT** – The entry conditions for AOP-004 include “toxic gas which impacts habitability of the Control Room with no impact on habitability in the power block outside the Control Room.” In this case, the habitability concerns are in the Aux Building outside the Control Room, so entry into AOP-004 and the actions to manually trip the reactor and main turbine / evacuate the Control Room are not applicable.

Technical Reference(s): AOP-004, Revision 23, “Control Room Inaccessibility”
OMM-001-2, Revision 85, “Shift Routines and Operating Practices”

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: fundamental knowledge or memory

10 CFR Part 55 Content: 55.41 (10)

K/A match: The K/A is matched since the applicant is required to assess plant conditions and make a determination of the applicable required actions per the guidance contained in their Conduct of Operations procedures (in this case, OMM-001-2 “Shift Routines and Operating Practices”).

QUESTION 62
034A4.01

K/A number and description:

034A4.01

034 Fuel Handling Equipment System (FHES)
A4 Ability to manually operate and/or monitor in the control room:
A4.01 Radiation levels

Question:

Given the following plant conditions,

- Unit 1 is in Mode 6 and refueling activities are in progress with the equipment hatch removed
- R-14C, Plant Vent Low Range Noble Gas Monitor is out of service.
- A fuel assembly has been visibly damaged during removal from the core.
- AOP-013, "Fuel Handling Accident", is in progress and CV rad levels are rising.

Radiation Monitor ____(1)_____ would provide the FIRST indications in the control room of a developing leak from the damaged fuel assembly. In accordance with AOP-013 the containment purge system is verified _____(2)_______ service.

R-2: CV area rad monitor
R-12: CV Air & Plant Vent Radioactive Gas

	(4)	(2)
M.	R-2	in
N.	R-2	NOT in
O.	R-12	in
P.	R-12	NOT in

Distractor Analysis:

Proposed Answer: C

A. Incorrect. The first part is incorrect, plausible since R-2 is a containment area rad monitor; however, its setpoint of 100 mR/hr and scaling will lag behind R-12 for initial indication with its scaling in dpm and a nominal setpoint at 1.8x background.

While R-2 may eventually show some increase, the activity released from a damaged fuel assembly is radioactive gases which is more readily detected by the gaseous monitor. The second part is correct; with the equipment hatch open during refueling operations, G-010 requires the R-11/12 in service with purge automatic isolation function defeated. With CV radiation levels rising AOP-013 will verify purge in service after checking the equipment hatch removed.

B. Incorrect. The first part is incorrect, plausible since R-2 is a containment area rad monitor; however, its setpoint of 100 mR/hr and scaling will lag behind R-12 for initial indication with its scaling in dpm and a nominal setpoint at 1.8x background. While R-2 may eventually show some increase, the activity released from a damaged fuel assembly is radioactive gases which is more readily detected by the gaseous monitor. The second part is incorrect; with the equipment hatch open during refueling operations, G-010 requires the R-11/12 in service with purge automatic isolation function defeated. With CV radiation levels rising AOP-013 will verify purge in service after checking the equipment hatch removed. Plausible because AOP-013 would verify purge out of service if the equipment hatch was closed per Mode 6 refueling requirements.

C. CORRECT. The first part is correct, The activity released from a damaged fuel assembly is radioactive gases which is more readily detected by the gaseous monitor and with a nominal setpoint of 1.8x background. The second part is correct; with the equipment hatch open during refueling operations, G-010 requires the R-11/12 in service with purge automatic isolation function defeated. With CV radiation levels rising AOP-013 will verify purge in service after checking the equipment hatch removed.

D. Incorrect. The first part is correct, The activity released from a damaged fuel assembly is radioactive gases which is more readily detected by the gaseous monitor and with a nominal setpoint of 1.8x background. The second part is incorrect; with the equipment hatch open during refueling operations, G-010 requires the R-11/12 in service with purge automatic isolation function defeated. With CV radiation levels rising AOP-013 will verify purge in service after checking the equipment hatch removed. Plausible because AOP-013 would verify purge out of service if the equipment hatch was closed per Mode 6 refueling requirements.

Reference(s)

- Annunciator Panel Procedure: APP-005, NIS and Reactor Control
- Abnormal Operating Procedure: AOP-005, RADIATION MONITORING SYSTEM, R30
- Abnormal Operating Procedure: AOP-013, FUEL HANDLING ACCIDENT, R15
- GP-010, Refueling
- FHP-001

K/A Match discussion:

The K/A is matched because the question involves conditions related to monitoring rad levels from the control room in response to a fuel handling event and taking appropriate actions.

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

Question Source: Bank: ______
Modified Bank: ______ (Note changes or attach parent)
New: ___X__

SRO Only Basis: N/A

QUESTION 63
035A3.01

035A3.01
Steam Generator
Ability to monitor automatic operation of the S/G including: S/G water level control
RO 4.0/SRO 3.9
CFR 41.7

A startup following a refueling outage is in progress with the reactor at 25% power and stable, holding for Chemistry with the following conditions:

- Rod control is in AUTO
- All other systems are in automatic

A 100 MWe load rejection occurs. Which one of the following describes the **overall** response and final condition for Steam Generator water level?

With no operator action, NR S/G water levels will ___________ .

A. decrease, then return to original level

B. increase, then return to original level

C. increase, then stabilize at a level lower than the original level

D. decrease, then stabilize at a level lower than the original level

Distractor Analysis

A. Incorrect. Answer is plausible due to being partially correct. Using 2339 MWt (790 MWe) as reactor core output at 100% power, 100 MWe equates to approximately 10% power. Reactor power will stabilize at a point below 20% power, therefore SGWL will stabilize on the 39-52% ramp following the transient. (Ref T.S., Page 1.1-4 and ST-027, Page 17)

B. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “A” and “C” distractor analyses

C. Incorrect. Answer is plausible due to being partially correct. SGWL responds to the transient by lowering (shrink phenomena). (Ref ST-048, Page 55)

D. Correct. Given information provides applicant success path.

References

- ST-027, Revision 4, Page 17
- ST-048, Revision 3, Page 55
- Technical Specifications, Amendment 196, Page 1.1-4

KA Match

Applicant is given a situation where system response evaluation of transient condition is required to arrive at correct conclusion.

Cognitive Level

High

Source of Question

Modified, NRC Bank

SRO Only Basis

Not applicable

QUESTION 64
068K5.04

068 K5.04
Liquid Radwaste System
Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: biological hazards of radiation and the resulting goal of ALARA.
3.2/3.5 (41.5, 45.7)

Which ONE of the following actions associated with transferring spent resin from the Spent Resin Storage Tank (SRST) to a High Integrity Container (HIC) is specified by OP-704, Spent Resin Storage Tank, to minimize radiation exposure to site personnel NOT involved in the transfer?

A. A PA announcement is made for unnecessary personnel to stay clear of the Auxiliary Building and the Radwaste Building.

B. The resin in the SRST is verified to have aged for 30 days, so there will be no radiation increases above normal Radiation Area limits during its transfer.

C. Signs are placed on all doors entering the Auxiliary Building and the Radwaste Building warning personnel to not enter except in case of an emergency.

D. Health Physics personnel with survey meters are stationed at key locations along the transfer route to prohibit personnel access.

Distractor Analysis

A. **CORRECT**. Reference OP-704 Step 8.3.2.4.

B. Incorrect. Plausible because there are time restrictions for releasing certain other radioactive effluents, such as Waste Gas Decay Tanks and Waste Condensate Tanks.

C. Incorrect. Plausible because it would have the same effect as making a PA announcement, and would in fact be more likely to stop someone who didn't hear the announcement, or who forgot.

D. Incorrect. Plausible because HPP-259, Spent Resin Transfer to Waste Processing Containers, is implemented by Health Physics in conjunction with the Operations transfer procedure, and Step 8.6.1.4 says, "**CONTROL** the traffic through the resin transfer areas.", and is an RC [Radiation Control] signoff step. The procedure doesn't describe *how* to accomplish that, and the question stem specifies OP-704, so D cannot be a correct answer.

References:
OP-704, Spent Resin Storage Tank
HPP-259, Spent Resin Transfer to Waste Processing Containers

KA Match:
The KA is matched because an applicant must recall what precaution is taken during a resin transfer to minimize dose to site personnel.

Cognitive Level: **Low**

Source of Question: **New**

QUESTION 65
079K1.01

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	K1.01	
Importance Rating	3.0	

Proposed Question:

Given the following plant conditions:

- The plant is operating at 100% RTP.
- The instrument air / service air cross connect bypass filter is tagged out for replacement.
- A loss of Instrument Air has occurred with pressure currently at 75 psig.
- The crew is implementing AOP-017, Loss of Instrument Air.

As pressure decreased, at __(1)__ psig Instrument Air header pressure, AOP-017 directed an operator to be dispatched to __(2)__.

Which one of the following completes the statement above?

A	(1) 85 (2) Open SA-70 (PRIMARY AIR COMP RECEIVER TO STATION AIR HEADER)
B	(1) 80 (2) Open SA-70 (PRIMARY AIR COMP RECEIVER TO STATION AIR HEADER)
C	(1) 85 (2) Open SA-5 (STATION AIR TO INST AIR CROSS CONNECT)
D	(1) 80 (2) Open SA-5 (STATION AIR TO INST AIR CROSS CONNECT)

Proposed Answer: **D**

K/A Match Analysis
079 Station Air System (SAS), K1.01
Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: Instrument Air System (IAS).

Requires knowledge of the physical connections and the proper use of those connections between SAS and IAS.

Answer Choice Analysis

A. (1) INCORRECT. Plausible because 85 psig is the entry condition for AOP-17
(2) INCORRECT. Plausible because SA-70 does cross connect IA and SA systems, but

is the lineup used to supply SA from IA.

B. (1) CORRECT. AOP-17 directs dispatching an operator to cross connect IA and SA at 80 psig.
   (2) INCORRECT.
C. (1) INCORRECT.
   (2) CORRECT. AOP-17 directs SA-5 to be opened if the coalescing filter is not available.
D. (1) CORRECT.
   (2) CORRECT.

Technical Reference(s):
ST-017 INSTRUMENT AND STATION AIR SYSTEM, Revision 3
AOP -017 LOSS OF INSTRUMENT AIR, REVISION 40

Proposed references to be provided to applicants during examination: None

Learning Objective:
Lesson AOP-017, rev. 40 Objective 3: Given a set of plant conditions, DETERMINE the appropriate plant response and operator actions in accordance with AOP-017.

Question Source: New

Question History: New

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Comments:

Supporting References
Attached

QUESTION 66
G2.1.1

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	G2.1.1
	Importance Rating	3.8

G2.1.1 – Knowledge of conduct of operations requirements

Proposed Question:

The plant is operating at 100% power when the following events occur:

- Shift personnel in the Control Room Complex begin to cough and their eyes start to burn concurrent with identification of an unusual odor
- A Unit Operator reports that a large chemical spill has occurred in the Auxiliary Building and recommends that the building be evacuated due to the resultant eye and respiratory irritation being experienced in the area

Based on the above conditions, the Control Room operating crew should perform which of the following actions?

5. Immediately don SCBAs
6. Align the Control Room Complex ventilation system to Emergency Recirculation
7. Manually trip the reactor and main turbine
8. Report to the Old Fire Equipment Building

a. 1 **only**

b. 2 **only**

c. 1 **and** 2

d. 3 **and** 4

Proposed Answer:

c. 1 **and** 2

Explanation:

This question requires the applicant to assess plant conditions and determine the required actions for a toxic gas event in the control room complex that does not require control room evacuation per the requirements of AOP-004 “Control Room Inaccessibility”.

a. **INCORRECT** – The applicant should determine that the unusual odor in the Control Room Complex coupled with the report of a chemical spill in the Auxiliary Room does not meet the entry conditions for AOP-004. In this case the applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation. Only donning SCBAs does not meet the requirements of the procedure.
b. **INCORRECT** – The applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation. Only donning SCBAs does not meet the requirements of the procedure. Only re-aligning Control Room ventilation does not meet the requirements of the procedure.
c. **CORRECT** – The applicant should determine that the unusual odor in the Control Room Complex coupled with the report of a chemical spill in the Auxiliary Room does not meet the entry conditions for AOP-004. In this case the applicant should determine that the applicable actions are from OMM-001-2 and thus SCBAs should immediately be donned in the Control Room **AND** Control Room Complex ventilation should be aligned for Emergency Recirculation.
d. **INCORRECT** – The entry conditions for AOP-004 include “toxic gas which impacts habitability of the Control Room with no impact on habitability in the power block outside the Control Room.” In this case, the habitability concerns are in the Aux Building outside the Control Room, so entry into AOP-004 and the actions to manually trip the reactor and main turbine / evacuate the Control Room are not applicable.

Technical Reference(s): AOP-004, Revision 23, “Control Room Inaccessibility”
OMM-001-2, Revision 85, “Shift Routines and Operating Practices”

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: fundamental knowledge or memory

10 CFR Part 55 Content: 55.41 (10)

K/A match: The K/A is matched since the applicant is required to assess plant conditions and make a determination of the applicable required actions per the guidance contained in their Conduct of Operations procedures (in this case, OMM-001-2 “Shift Routines and Operating Practices”).

QUESTION 67
G2.1.27

G2.1	Conduct of Operations
G2.1.27	Knowledge of system purpose and/or function.

Question:

Which one of the following includes acceptance criteria for the ECCS following a postulated Loss-of-Coolant Accident, as required by 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear reactors?"

The calculated maximum fuel element cladding temperature shall be maintained less than _____(1)_____ degrees F and the maximum cladding oxidation less than _____(2)_____ percent.

	(5)	(2)
Q.	2200	1
R.	2200	17
S.	2500	1
T.	2500	17

Distractor Analysis:

Proposed Answer: B

A. Incorrect. The first part is correct. The second part is incorrect; plausible because 1 % is the ECCS acceptance criteria for H2 production.

B. Correct.

C. Incorrect. The first part is incorrect, plausible because 2500 is near the limit but still well below zirconium melting temperature

D. Incorrect. The first part is incorrect, plausible because 2500 F is near the limit but still well below zirconium melting temperature. The second part is incorrect; plausible because 1 % is the ECCS acceptance criteria for H_2 production.

Reference(s)

- System Description SD-006, ESFAS
- Student Test SD-006 Objectives.

K/A Match discussion:

The K/A is matched because the question involves knowledge of the ESFAS system purpose.

Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

Question Source:	Bank:	X	
	Modified Bank:		(Note changes or attach parent)
	New:		

Source was LXR test bank for Westinghouse 3-Loop.

Only information provided was:

REF: SD 021/SYS 050, 062, 064 2/6/98, page 10. 10 CFR 50.46

SRO Only Basis: N/A

QUESTION 68
G2.2.2

Equipment Control
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
RO 4.6/SRO 4.1
CFR 41.6/41.7

The operating crew is performing actions to place the plant in Mode 5 in accordance with GP-007, "PLANT COOLDOWN FROM HOT SHUTDOWN TO COLD SHUTDOWN" with the following sequence of events:

- 0830 MDAFW Pump 'B' placed in service
- 1430 MDAFW Pump 'B' secured

- 1432 RHR Pump 'A' placed in service
- 1448 RHR Pump 'A' secured due to a flow indication problem on FI-608
- 1458 RHR Pump 'A' placed in service
- 1505 RHR Pump 'A' secured due to a flow indication problem on FI-608
- 1509 RHR Pump 'A' placed in service
- 1525 RHR Pump 'A' secured due to a flow indication problem on FI-608

Given: FI-608, "RHR SYSTEM HEATUP LINE FLOW INDICATION"

The time is currently 1530

The maximum number of consecutive pump starts available for MDAFW Pump 'B' in accordance with OP-402, "AUXILIARY FEEDWATER SYSTEM" is ____(1)____ .

A 45 minute waiting period prior to restarting RHR Pump 'A' ____(2)____ required in accordance with OP-201, "RESIDUAL HEAT REMOVAL SYSTEM".

	(1)	(2)
A.	2	Is
B.	2	Is NOT
C.	3	Is
D.	3	Is NOT

Distractor Analysis

A. Incorrect. Plausible if applicant misapplies incorrect answers to question, see “B” and “C” distractor analyses.

B. Incorrect. Answer is plausible due to being partially correct. Since MDAFW ‘B’ has not been run in the last hour, three consecutive pump starts are permitted. Although the two consecutive pump start criteria is also met, a total of three starts (with the given conditions) are available. (Ref OP-402, Page 12)

C. Incorrect. Answer is plausible due to being partially correct. A 45 minute waiting period would be required if the last RHR ‘A’ run had been less than 15 minutes as the pump has been run three times within the last 45 minutes (per the given conditions). (Ref OP-201, Page 9)

D. Correct. Given information provides applicant success path.

References

- OP-402, Revision 86, Page 12
- OP-201, Revision 69, Page 9

KA Match

Applicant is given a situation where evaluation of running conditions is required to arrive at correct conclusion.

Cognitive Level

High

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 69
G2.2.42

2.2.42
Ability to recognize system parameters that are entry-level conditions for Tech Specs.
3.9/4.6
(41.7, 41.10, 43.2, 43.3, 45.3)

Given the following:

- The plant is at 100% power.
- A malfunction in LC-459G, PRESSURIZER LEVEL, has caused PZR level to increase to 64% over the last 10 minutes. The controller remains in AUTO, with no operator action.
- Pressurizer heaters and sprays over-compensated for the rising pressure caused by rising level, and stabilized PZR pressure at 2205 psig by the lowest indicator.
- All other RCS parameters are within their normal control bands.

The Reactor Operator should notify the CRSS that:

A. the reactor should be tripped because a trip setpoint will be reached within the next 2 minutes.

B. an entry condition for ITS 3.4.9, Pressurizer, has been met.

C. an entry condition for ITS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, has been met.

D. entry into AOP-025, RTGB Instrument Failure, is required.

Distractor Analysis

A. Incorrect. Program PZR level at 100% power is 53.3%, so it's up almost 10.7% in 10 minutes, about 1% per minute. 75% PZR level will trip the reactor, but that won't be reached for another (75 – 64) *
1 min/% = 11 minutes. 5x longer than the time in the distractor.

B. **CORRECT**. LCO 3.4.9 says that PZR shall be <63% in Mode 1. This is above-the-line information.

C. Incorrect. LCO 3.4.1 states that "RCS DNB parameters ... shall be within ... limits: a. Pressurizer pressure ≥ 2205 psig." The stem puts them exactly at that point, to test whether they'll remember that it's greater than *or equal to*, or just greater than. It's somewhat minutia, but it was difficult to find another plausible distractor, so this makes this distractor pretty plausible.

D. Incorrect. Most instrument-related PZR transients put you in AOP-025, RTGB Instrument Failure. In this question they're specifically told it's a level controller problem (not a level *instrument*) problem, which is not an entry condition for the AOP.

References:

AOP-035, RTGB Instrument Failure, Rev. 20.
ITS LCO 3.4.1
ITS LCO 3.4.9

KA Match:

The KA is matched because the applicant must be able to recognize that current PZR level is an entry condition for a 6-hour shutdown tech spec.

Cognitive Level: **Low**

Source of Question: **New**

QUESTION 70
G2.3.11

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G 2.3.11	
	Importance Rating	3.8	

Proposed Question:

A release of a Waste Gas Decay Tank ___(1)___ allowed while simultaneously releasing the containment via a continuous CV purge.

A release of a Waste Gas Decay Tank is allowed while stack monitor R-14C is out of service ___(2)___

Which ONE of the following completes the statements above?

A	1) is 2) as long as R-14D is operable because the ranges overlap.
B	1) is NOT 2) as long as R-14D is operable because the ranges overlap.
C	1) is 2) as long as an independent sample of the tank to be released has been performed.
D	1) is NOT 2) as long as an independent sample of the tank to be released has been performed.

Proposed Answer: **D**

K/A Match Analysis
G 2.3.11 Radiation Control, ability to control radiation releases.

Requires the ability to control a Waste Gas Decay Tank radioactive materials release.

Answer Choice Analysis

A. INCORRECT. First part is correct. A WGDT release may be conducted while CV continuous purge is in progress. Second part is plausible because R-14C and R-14D ranges do overlap and both monitor the exhaust stack for noble gas.

B. INCORRECT. First part is plausible because a tank release may NOT be conducted if a CV batch release is in progress. Second part is also incorrect.

C. CORRECT. First part is correct. Second part is correct, a tank release permit can be issued if a second, independent sample is performed.

D. INCORRECT.

Technical Reference(s):

EMP-022, GASEOUS WASTE RELEASE PERMITS, Revision 54
OP-706, WASTE DISPOSAL-GASEOUS RAD WASTE RELEASE, Revision18
OP-921, CONTAINMENT AIR HANDLING, Revision 53
ST-019, RADIATION MONITORING SYSTEM, Revision 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:	New	
Question History:	New	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	_
10 CFR Part 55 Content:	55.41 X	
	55.43 _	

Comments:

Supporting References

Attached

QUESTION 71
G2.3.5

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	G2.3.5
	Importance Rating	2.9

G2.3.5 – Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question:

You have been directed by the Control Room Supervisor to perform a source check of Area Radiation Monitor R-32A (CV High Range) per OP-920, "Radiation Monitoring System."

Prior to depressing the ECS pushbutton on R-32A you should ensure the ratemeter Selector Switch is in the ____(1)____ position.

Proper response of R-32A is verified by ____(2)____.

	(1)	(2)
a.	ALL	Indication of ~10^3 R/H on the analog display
b.	ALL	ALERT, HIGH, and CHANNEL TEST lamps energized
c.	TEST	ALERT, HIGH, and CHANNEL TEST lamps energized
d.	TEST	Indication of ~10^3 R/H on the analog display

Proposed Answer:

a. ALL Indication of ~10^3 R/H on the analog display

Explanation:

This question requires the applicant to recall the proper methodology for performing a source check on a containment vessel high range radiation monitor.

a. **CORRECT** – When the ECS button is depressed and held with the ratemeter selector switch for R-32A in the ALL position, an electrical current which simulates a radiation source is supplied to the detector. Proper ratemeter operation is verified by a reading of about 10^3 R/H and the green SAFE/RESET light staying on. The process will not result in either an ALERT or HIGH rad alarm.
b. **INCORRECT** – When the ECS button is depressed and held with the ratemeter selector switch for R-32A in the ALL position, an electrical current which simulates a radiation source is supplied to the detector. Proper ratemeter operation is verified by a reading of about 10^3 R/H and the green SAFE/RESET light staying on. The process will not result in either an ALERT or HIGH rad alarm. A channel test of R-32A will result in the indications stated in the distractor, thus making it plausible.
c. **INCORRECT** – This distractor states the ratemeter selector switch position and indications that would be received during a channel test of R-32A, NOT a source check. Plausible if the applicant confuses source check indications with a channel test.
d. **INCORRECT** – This distractor is only half-correct. The indications for proper operation are correctly stated, but the position of the ratemeter selector switch should be ALL (not TEST).

Technical Reference(s): ST-019, "Radiation Monitoring System"
OP-920, Revision 40 "Radiation Monitoring System"

Proposed references to be provided to applicants during examination: none

Question Source: new

Question History: none

Question Cognitive Level: memory or fundamental knowledge

10 CFR Part 55 Content: 55.41 (11)

K/A match: The K/A is matched since the applicant is expected to demonstrate knowledge of how to properly perform (and the expected response) a source check of a radiation monitor (R-32A).

QUESTION 72
G2.3.7

K/A number and description:

G2.3.7

G2.3 Radiation Control
G2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question:

Given the following:

- While entering the plant to relieve the shift, you notice smoke coming from the Reactor Auxiliary Building.

- You have NOT logged onto any RWP or obtained proper dosimetry (EAD).

- Your assistance is required in the Auxiliary Building.

Which one of the following is allowed/required IAW FP-001, FIRE EMERGENCY for the stated conditions?

A. An exception to the requirement for an EAD AND RWP login is allowed for qualified members of the Fire Brigade (No RC Tech is required to invoke the exception).

B. An EAD must be obtained AND RWP login must be completed prior to Auxiliary Building entry.

C. An exception to the requirement for obtaining an EAD AND RWP login is allowed ONLY IF an RC Tech is present to monitor personnel entry/exit to the Auxiliary Building.

D. RWP login is NOT required, and emergency dosimetry (EAD) is obtained from the RCA entrance.

Distractor Analysis:

Proposed Answer: B

A. Incorrect. Exception for dosimetry and RWP login is NOT allowed. Operations personnel are required to obtain proper dosimetry prior to relieving their watchstations.

B. Correct. RWP login and proper dosimetry is required prior to entry into any RCA.

C. Incorrect. RC tech being present does NOT relieve personnel of the requirement for proper dosimetry.

D. Incorrect. Emergency dosimetry is NOT available in the RCA entrance.

Reference(s)

- OMM-001-12, Minimum Equipment List and Shift Relief, Rev 75
- Fire Emergency, FP-001, Rev 63

K/A Match discussion:

The K/A is matched because the question involves radiation work permit requirements during normal or abnormal conditions.

Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

Question Source:	Bank:	X	
	Modified Bank:		(Note changes or attach parent)
	New:		

HLC -08 NRC Exam.

SRO Only Basis: N/A

QUESTION 73
G2.4.1

Question 73
G2.4.1
Emergency Procedures/Plan
Knowledge of EOP entry conditions and immediate action steps.
RO 4.6/SRO 4.8
CFR 41.10/43.5

A reactor shutdown is in progress in accordance with GP-006-1, “NORMAL PLANT SHUTDOWN FROM POWER OPERATION”

Initial Conditions:

- Reactor Power is 3%
- The Main Turbine has been shutdown
- The electric plant is in a normal shutdown alignment with buses supplied from the SUT

Current Conditions:

- Operators have depressed both Reactor Trip pushbuttons in accordance with GP-006-1
- Reactor Trip breakers remain CLOSED
- Intermediate Range SUR indication is fluctuating between 0.0 and -0.1 DPM

Note: EOP-E-0, “REACTOR TRIP OR SAFETY INJECTION”
FRP-S.1, “RESPONSE TO NUCLEAR POWER GENERATION/ATWS”

Which answer below describes the required crew response to the above conditions?

A. Enter EOP-E-0, do NOT enter FRP-S.1

B. Enter FRP-S.1, do NOT enter EOP-E-0

C. Enter EOP-E-0 AND FRP-S.1

D. Do NOT enter EOP-E-0 OR FRP-S.1

Distractor Analysis

A. Correct. Given information provides applicant success path.

B. Incorrect. Plausible if applicant misapplies incorrect answers to question, see "C" and "D" distractor analyses.

C. Incorrect. Answer is plausible due to being partially correct. Per GP-006-1, entry into EOP-E-0 is not required due to tripping the reactor as part of a planned shutdown evolution. However, the procedure also stipulates that in the event of a failure to trip of the Reactor Trip breakers, entry into EOP-E-0 is required. Similar wording is presented in OMM-022. (Ref OMM-022, Page 10 and GP-006-1, Page 45)

D. Incorrect. Answer is plausible due to being partially correct. Entry into FRP-S.1 is not required per EOP-E-0 immediate action step 1 (neither the reactor power nor IR SUR limits are satisfied in the question stem). OMM-022 states that FRP-S.1 entry is required, but no procedural entry condition is met. (Ref EOP-E-0, Page 6; FRP-S.1, Page 2; OMM-022, Page 10; & CSFST, Page 3)

References

- OMM-022, Revision 43, Page 10
- GP-006-1, Revision 9, Page 45
- EOP-E-0, Revision 5, Page 6
- FRP-S.1, Revision 20, Page 2
- CSFST, Revision 6, Page 3

KA Match

Applicant is given a situation where evaluation of plant conditions and procedural entry requirements is required to arrive at correct conclusion.

Cognitive Level

High

Source of Question

New

SRO Only Basis

Not applicable

QUESTION 74
G2.4.32

G2.4.32
Knowledge of operator response to loss of all annunciators.
3.6/4.0 (41.10, 43.5, 45.13)

Given the following:

- The plant is at 100% power.
- Control Room receives APP-036-H5, ANNUN SYS DC PWR LOST.
- The plant remains stable.

As Balance of Plant Operator, you direct the Turbine Building Auxiliary Operator to check Breaker 10, Annunciator Panel (RTGB), on Distribution Panel **__(1)__.**

Per OP-603, Electrical Distribution, Precautions & Limitations, if there is no evidence of abnormality present, the breaker can be reset **__(2)__**.

	(1)	(2)
A.	A	one time
B.	B	only after consultation with Engineering
C.	A	only after consultation with Engineering
D.	B	one time

Correct answer: D

Distractor Analysis

RTGB Annunciators are fed by Distribution Panel 'B' breaker 10, straight from 'B' Battery. The only other safety-related battery is 'A', so Distribution Panel 'A' is just as plausible.

OP-603 P&L 5.5.4 allows resetting a protection device with CRS permission *one time*. P&L 5.6 says the SM may approve *additional* resets after consultation with engineering.

References:
APP-036-H5, Rev. 85
OP-603, Electrical Distribution, Rev. 118

KA Match:
The KA is matched because applicant must demonstrate knowledge of operator response to loss of all annunciators.

Cognitive Level: **Low**

Source of Question: **New**

QUESTION 75
G2.4.4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G 2.4.4	
	Importance Rating	3.8	

Proposed Question:

Given the following plant conditions:

- The unit is operating at 100% power
- Grid frequency is beginning to lower

The highest frequency, below which entry into AOP-026 (Grid Instability) will be required, is (1) .

The highest frequency, below which an automatic Reactor trip will occur, is ___(2)___.

Which one of the following completes the statements above?

A	(1) 59.8 Hz (2) 58.2 Hz
B	(1) 59.8 Hz (2) 58.4 Hz
C	(1) 59.0 Hz (2) 58.2 Hz
D	(1) 59.0 Hz (2) 58.4 Hz

Proposed Answer: **A**

K/A Match Analysis

G 2.4.4 Emergency Procedures / Plan, ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Requires the ability to recognize abnormal generator frequencies that are entry-level conditions for AOP-026 and the EOP-E-0.

Answer Choice Analysis

U. (1) CORRECT. One of the entry conditions for AOP-026 is observed frequency less than 59.8 Hz.
(2) CORRECT. All reactor coolant pumps will automatically trip at 58.2Hz, which if above 10% power, will automatically trip the reactor.

V. (1) CORRECT. One of the entry conditions for AOP-026 is observed frequency less than 59.8 Hz.
(2) INCORRECT. Plausible because if less than 58.4 Hz, AOP-026 step 2 directs a manual reactor trip, trip of RCPs, and transition to EOP-E-0.

W. (1) INCORRECT. Plausible because of a five minute time limit on frequency between 58.4 and 59.0 Hz exists in AOP-026, if above 59.0 Hz the time limit no longer applies.
(2) CORRECT

X. (1) INCORRECT
(2) INCORRECT

Technical Reference(s):

AOP-026, Grid Instability, Rev 14
ST-011, Reactor Protection System, , Rev
Proposed references to be provided to applicants during examination: None

Learning Objective:

Lesson AOP-026-OIT rev. 13:
Objective 1: Given a set of plant conditions, determine when entry into AOP-026 is appropriate.
Objective 2: STATE/IDENTIFY the following for AOP-026:
a. Conditions requiring a reactor trip.
b. Conditions requiring a trip of the RCPs.

Question Source: New

Question History: New

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

10 CFR Part 55 Content: 55.41 X
55.43 _

Supporting References

Attached