Examination Outline Cross-reference:

(Page 10) 205000 Shutdown Cooling K2.02

Knowledge of electrical power supplies to the following:

- Motor operated valves

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	205000 K2.02	
Importance Rating	2.5	

Proposed Question: **# 1**

Currently, 10MOV-18 (RHR Shutdown Cooling Suction Isolation Valve) is "OPEN" with the associated Appendix R Power Disconnect and Isolation Switches in the following positions:

- Power Disconnect Switch is in the "CLOSED" position
- Isolation Switch is in the "LOCAL" position

From which of the following locations can 10MOV-18 be CLOSED?

a. On the Valve Control Switch located on 10MOV-18
b. Control Room Panel 09-4 Valve Control Switch for 10MOV-18
c. BMCC-4 Valve Control Switch located on the Motor Controller Front Panel
d. MCC-156 Valve Control Switch located on the Motor Controller Front Panel

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: with the isolation switch in "LOCAL", 10MOV-18 may be operated electrically from MCC-156 using the valve control switch located on the motor controller – allows closure from a convenient location and provides a location for isolation switch that would not be compromised during a LOCA due to radiological conditions.

a. Incorrect: 10MOV-18 is in the Drywell, but has no valve control switch mounted on it.

b. Incorrect: with the isolation switch in "LOCAL", 10MOV-18 may not be operated from the Control Room. The switch must be in "REMOTE" to allow Control Room operation of 10MOV-18 from the 09-4 Panel.

c. Incorrect: correct concept, but power supply is for 10MOV-17 (RHR Shutdown Cooling Suction Valve) and 10MOV-17 is not set up in this manner.

Technical Reference(s): OP-13D (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO 1.11.b.2 (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 20) 268000 Radwaste K1.03

Knowledge of the physical connections and/or cause-effect relationships between RADWASTE and the following:

- Reactor building equipment drains: Plant-Specific

Level	RO
Tier #	2
Group #	2
K/A #	268000 K1.03
Importance Rating	2.6

Proposed Question: **# 2**

Because leakage collected in the Reactor Building Equipment Drain Sumps is considered to be _____, this volume is pumped directly to either the _____ or the _____.

a. Low Conductivity (High Purity) / Waste Collector Tank *or the* Waste Surge Tank
b. High Conductivity (Low Purity) / Floor Drain Collector Tank *or the* Waste Surge Tank
c. Low Conductivity (High Purity) / Waste Sample Tanks *or the* Condensate Storage Tanks
d. High Conductivity (Low Purity) / Waste Sample Tanks *or the* Radwaste Equipment Drain Sump

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – Low Conductivity and the WCT is where that is primarily delivered with the Waste Surge Tank being a surge volume
b. If the improper assumption is made for whatever reason, that the drains are high conductivity, then these are potential delivery locations.
c. Although it is considered Low Conductivity Waste, it would not be sent directly to the CST's at a minimum and not the WST's without some filtering/processing
d. If the improper assumption is made for whatever reason, that the drains are high conductivity, then these are potential delivery locations.

Technical Reference(s): OP-49 (Attach if not previously provided)

FM-17A thru C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-20, EO 1.05.a & 1.06 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 22) G2.1.20
Ability to execute procedure steps.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	Cat 1 G2.1.20	
Importance Rating	4.3	

Proposed Question: **# 3**

With the EPIC computer system down for reboot, a logic failure has resulted in a dual recirculation pump runback. Several minutes after the runback, the following indications were observed:

- 09-5 Average APRM Power is 62%
- 09-5 Total Core Flow is 34 Mlbm/hr

Which of the below describes the required operator actions?

a. Insert Control Rods to reduce reactor power by at least 5%.
b. Raise recirculation flow to raise reactor reactor power by at least 5%.
c. Insert Control Rods to reduce reactor power by at least 30%.
d. No action required, operation in this conditions is acceptable.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: These conditions place operation in the unshaded buffer region of RAP-7.3.16, Att 4/5. In this region with Soloman unavailable, AOP-8 requires exit by control rod insertion or raising of recirculation flow. The stem conditions prohibit the raising of recirculation flow.

b. Incorrect: Stem conditions prohibit the raising of recirculation flow

c. Incorrect: Incorrect use of the RAP-7.3.16 Att 4/5 curve would show operation in the Shaded Exclusion region if the lower axis is read on the % Core Flow values rather than the mlbm/hr values. This act would require a power reduction by at least 30%.

d. Incorrect: Misreading the stem conditions as Solomon being available would make this an acceptable answer.

Technical Reference(s): AOP-8 (Attach if not previously provided)

RAP-7.3.16

Proposed references to be provided to applicants during examination: AOP-8, RAP-7.3.16, Att 4 or 5

Learning Objective: LP-AOP EO 2.08 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 22) G2.1.27
Knowledge of system purpose and/or function.

Level	RO	[redacted]
Tier #	3	
Group #	[redacted]	
K/A #	Cat 1 G2.1.27	
Importance Rating	2.8	[redacted]

Proposed Question: **# 4**

Reactor Power is reduced from 100% to 95% by lowering recirculation flow.

Turbine Control Valves are repositioned by EHC sensing _____ as compared to _____.

a. RPV Pressure; Pressure Setpoint
b. RPV Pressure; Turbine 1st Stage Pressure
c. Turbine Inlet Pressure; Pressure Setpoint
d. Turbine Inlet Pressure; Turbine 1st Stage Pressure

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT – A recirc flow reduction will result in lower core thermal power and, therefore, lower steam pressure. The lower steam pressure sensed at the Pressure Averaging Manifold will result in further control valve closure; thus lowering MWe. The Pressure Control Unit is a proportional controller which positions control valves proportional to the error between pressure set and actual pressure.

a. Incorrect – See above.

b. Incorrect – See above.

d. Incorrect – See above.

Technical Reference(s): SDLP-94C (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94C, EO 1.05.a.4 (As available)

Question Source: Bank # 25679

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 12) 215004 Source Range Monitor K4.04

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following:

- Changing detector position

Level	RO
Tier #	2
Group #	1
K/A #	215004 K4.04
Importance Rating	2.8

Proposed Question: **# 7**

A reactor startup is in progress. "Alpha" and "Charlie" Source Range Monitors (SRMs) are to be withdrawn to maintain the count rate level band. The "RETRACT PERMIT" light is illuminated (on) for SRM "A" and is **NOT** illuminated (off) for SRM "C". The Reactor Operator selects SRMs "A" and "C" and presses the "DRIVE OUT / DRIVING OUT" pushbutton.

Which of the following occurs?

a. Neither of the selected SRMs moves from its full-in position and a control rod block occurs.

b. "Alpha" SRM moves outward and "Charlie" SRM does NOT move due to a SRM withdraw block.

c. Both of the selected SRMs will continue to move outward until either SRM withdraw block occurs.

d. Both of the selected SRMs will continue to move outward and a concurrent control rod block occurs.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Backlighting behind upper half of information window (Green) "RETRACT PERMIT" indicates detector may be withdrawn without causing a rod block. This is For Information Only, as there are no interlocks to inhibit detector insertion or withdrawal.

a. Incorrect – See above

b. Incorrect – See above

c. Incorrect – See above

Technical Reference(s): ARP-09-5-2-2 / ARP-09-5-2-21 (Attach if not previously provided)

OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.11.b.4 (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
(Page 2) 295001 Partial or Complete Loss of Forced Core Flow Circulation AA1.07	Tier #	1	
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :	Group #	1	
	K/A #	295001 AA1.07	
• Nuclear Boiler Instrumentation System	Importance Rating	3.1	

Proposed Question: **# 8**

The "Bravo" Recirculation Pump trips during normal, full-power operation. If all the jet pumps are intact and operating properly, you would expect to see which of the following?

When the "B" Recirculation Pump is trips, the flow indication for all the "Bravo-loop" Jet Pumps should decrease to zero as the pump coasts to a stop. Then, the flow indication for all of the "B-loop" Jet Pumps should _____ and the flow indication for all of the "Alpha-loop" Jet Pumps should _____.

a. drop below zero as flow reverses; increase during the transient.
b. drop below zero as flow reverses; not change during the transient.
c. **increase to a positive value as flow reverses; increase during the transient.**
d. increase to a positive value as flow reverses; not change during the transient.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT – Per AOP-8, a rise in flow will result in the operating loop. RAP-7.3.25 graphically illustrates in Att. 1. Per UFSAR, Section 7.1.6.b, the discussion of the reason for the Reverse Flow Summer indicates the reasoning for removing the apparent positive flow through inactive Jet Pumps during single-loop operations.

a. Incorrect – The first part of the distracter is invalidated by the above UFSAR discussion. The second part is true, per the above discussion of AOP-8 and RAP-7.3.25.

b. Incorrect – Neither part of this distracter is true, per above references to AOP-8, RAP-7.3.25, and the UFSAR.

d. Incorrect – The first part of the distracter is true. The second part is invalidated by the above discussion of AOP-8 and RAP-7.3.25.

Technical Reference(s): Tech Spec 3.4.2 Bases, UFSAR 7.1.6.b (Attach if not previously provided)

RAP-7.3.25, AOP-8, AOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02E, EO 1.05.a.7/8 & 1.06.d (As available)

SDLP-02I, EO 1.09.d/e

Question Source: Bank # 20448 (Quad Cities 8/13/01)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 18) 201003 Control Rod and Drive Mechanism K6.02

Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM:

- Reactor pressure

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	201003 K6.02	
Importance Rating	3.0	

Proposed Question: **# 9**

During a reactor startup, a common-cause O-Ring failure has caused the operability of all 137 CRD Hydraulic Control Unit (HCU) Accumulators to be called into question. This would be of concern if reactor pressure were below _____ due to the fact that if a full scram signal occurred and the HCUs failed _____.

a. 800 psig; reactor pressure alone may not be sufficient to fully insert rods.
b. 800 psig; reactor pressure alone may not result in required scram times.
c. 1000 psig; reactor pressure alone may not be sufficient to fully insert rods.
d. 1000 psig; reactor pressure alone may not result in required scram times.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: Reference SDLP-03a, Figure 8 "Accumulator to Reactor Pressure Scram Times. 800 psig is the pressure below which reactor pressure alone will result in insufficient scram times to prevent exceeding fuel thermal limits during Design Basis Accidents and transients.

a. Incorrect: It is the correct pressure, but the wrong reason, as reactor pressure will always insert the control rod; it just may take a substantial amount of time to accomplish.

c. Incorrect: Incorrect pressure and wrong reason, as reactor pressure will always insert the control rod; it just may take a substantial amount of time to accomplish.

d. Incorrect: Incorrect pressure which causes the correct reasoning to be invalid.

Technical Reference(s): SDLP-03A, Figure 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03A, EO 1.10.b (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 3) 295018 Partial or Total Loss of CCW AK2.01

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following:

- System loads

Level	RO
Tier #	1
Group #	1
K/A #	295018 AK2.01
Importance Rating	3.3

Proposed Question: **# 10**

The plant is operating at 90% power with one Reactor Building Closed Loop Cooling (RBCLC) pump tagged out of service. An electrical problem causes the two running RBCLC pumps to trip.

Operators have the ability to restore cooling via Emergency Service Water to EACH of the following EXCEPT:

a. Drywell Cooling Assemblies
b. Drywell Equipment Sump Cooler
c. Recirculation Pump Seal Coolers
d. RWCU Non-Regenerative Heat Exchanger

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Per AOP-11 actions, operators must consider shutting down RWCU (as all cooling flow to the RWCU NRHX will be lost).

a. Incorrect – Per AOP-11 actions, ESW can/will be supplied to this load.

b. Incorrect – Per AOP-11 actions, ESW can/will be supplied to this load.

c. Incorrect – Per AOP-11 actions, ESW can/will be supplied to this load.

Technical Reference(s): AOP-11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15, EO 1.06.b & 1.09 (As available)

Question Source: Bank # 25604

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 2) 295004 Partial or Total Loss of DC Pwr AK2.03
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following:

- D.C. bus loads

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295004 AK2.03	
Importance Rating	3.3	

Proposed Question: **# 11**

A Loss of Coolant Accident has just occurred. Conditions are as follows:

- RPV water level is +58 inches
- RPV Pressure is 500 psig
- Annunciator 09-8-3-7, "LPCI MOV IPS A BATT VOLTS LO OR BKR TRIP" is in alarm
- NPO reports that the LPCI Battery Breaker 1CB2 is destroyed

Which of the following operator actions / event sequences will support successful "Alpha" Low Pressure Coolant Injection System (LPCI) injection to the reactor vessel?

a. NO immediate operator actions are necessary. Valves required for LPCI "A" operation will reposition, when required, using the normal AC feed (MCC-152).

b. Immediately take the LPCI MOV A PWR SUPP switch to "ISOLATE". Valves required for LPCI "A" operation will then reposition, when required, using the alternate AC feed (MCC-153).

c. **Immediately take the LPCI MOV A PWR SUPP switch to "ALT PULL TO LOCK". Valves required for LPCI "A" operation will then reposition, when required, using the alternate AC feed (MCC-153).**

d. After waiting a mandatory 10 minutes, take the LPCI MOV A PWR SUPP switch to "BYPASS". Valves required for LPCI "A" operation will then reposition, when required, using the normal AC feed (MCC-152).

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: On a loss of LPCI DC Power, Per ARP 09-8-3-7, "If LPCI Bus immediately required..." this is the only action directed.

a. Incorrect: Normal AC Feed would trip on LPCI initiation signal. Even so, with the DC lost/breaker open, the normal AC feed to the inverter/charger will not provide adequate current to the inverter to allow stroking any of the MOVs. (Reference OP-43C, Caution C.2.5)

b. Incorrect: Taking the switch to "ISOLATE" will result in immediate opening of the normal AC input breaker from MCC-152, but does not contain logic to swap to the alternate AC feed.

d. Incorrect: This plays on the 10 minute wait after a LPCI initiation signal, where taking the switch to "BYPASS" will allow the closing of the normal AC input breaker to the inverter/charger. See above.

Technical Reference(s): OP-43C (Attach if not previously provided)

ARP 09-8-3-7

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B, EO 1.05.b.1, 1.13.c, & 1.14.c.12 (As available)

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 4) 295023 Refueling Acc Cooling Mode AK3.02

Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS:

- Interlocks associated with fuel handling equipment

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295023 AK3.02	
Importance Rating	3.4	

Proposed Question: **# 12**

Fuel Handling operations are in progress. The following conditions exist:

- Mode switch in REFUEL
- Fuel Grapple NOT loaded
- Fuel Grapple full up
- One (1) control rod selected and fully withdrawn
- Bridge over the Spent Fuel Pool

From these conditions, which one (1) of the following restrictions occurs and why?

a. Bridge motion near or over the core will not be permitted, to prevent bridge operator overexposure

b. A second control rod selected will cause a rod block, to prevent bridge operator overexposure

c. Bridge motion near or over the core will not be permitted, to prevent inadvertent criticality

d. A second control rod selected will cause a rod block, to prevent inadvertent criticality

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – OP-66A, Attachments 2, 3, and 4 indicate via flowcharts what will result in a rod block, bridge motion block, and a bridge hoist block. Selecting the second control rod will, in fact cause a rod block, per Attachment 2. UFSAR Section 14.6.1.4 on Refueling Accidents. "The refueling interlocks, which impose restrictions on the movements of refueling equipment and control rods, prevent inadvertent criticality during refueling operations..."

a. Incorrect – Per Attachment 3, stem conditions will not cause a bridge motion block and reasoning would be invalid.

b. Incorrect – Although the selection of the second rod is true, the reasoning is invalid (see above).

c. Incorrect – Although the reasoning is true as it relates to Refuel Interlocks as a whole, the stem conditions will not cause a bridge motion block (per Attachment 3).

Technical Reference(s): OP-66A (Attach if not previously provided)

UFSAR 14.6.1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08B, EO 1.02 & 1.05 (As available)

Question Source: Bank # 25607

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 3) 295019 Partial or Total Loss of Inst. Air AA1.04

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

- Service air isolation valves: Plant-Specific

Level	RO
Tier #	1
Group #	1
K/A #	295019 AA1.04
Importance Rating	3.3

Proposed Question: **# 13**

The Instrument Air System (IAS) has developed a leak. Which of the following describes the operation of the Service Air Isolation Valve (39FCV-110) during this transient?

a. Automatically closes at 85 psig (lowering) and will automatically reset at 90 psig (rising)
b. Automatically closes at 85 psig (lowering) and can be manually reset at 90 psig (rising)
c. Automatically closes at 95 psig (lowering) and will automatically reset at 100 psig (rising)
d. Automatically closes at 95 psig (lowering) and can be manually reset at 100 psig (rising)

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Per AOP-12 and OP-39, 39FCV-110 will automatically close if air header pressure drops to 95 psig. Additionally, OP-39 specifies that the red Reset pushbutton must be manually pressed (which overrides and opens valve) and then released when header pressure is greater than 100 psig (to maintain valve open)

a. Incorrect: See above. 85 psig is the setpoint for the **Breathing** Air Header Isolation Valve 39AOV-111. Reset pressure is correct for 39AOV-111, but does **not** automatically reset.

b. Incorrect: See above. 85 psig is the setpoint for the **Breathing** Air Header Isolation Valve 39AOV-111. Function and Reset pressure is correct for 39AOV-111.

c. Incorrect: See above. Correct setpoint for Isolate and Reset, but does **not** automatically reset.

Technical Reference(s): AOP-12 (Attach if not previously provided)

OP-39, Section G.11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39, EO 1.05.a.9/10, 1.05.c.1/2, & 1.11.b.3/4 (As available)

Question Source: [redacted] Bank # [redacted]

Modified Bank # 24638 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
Seabrook 1	5/30/2003	ILO	24638	PWR

Which of the following describes the operation of the Service Air isolation valves, SA-V92 and SA-V93, during an Instrument Air leak?

Answer **Automatically CLOSE at 90 psig decreasing, resets to allow manual opening above 95 psig increasing.**

Distractor 2 Automatically CLOSE at 80 psig decreasing, automatically REOPEN above 83 psig INCREASING.

Distractor 3 Automatically CLOSE at 90 psig decreasing, automatically reopen above 95 psig INCREASING.

Distractor 1 Automatically CLOSE at 80 psig decreasing, resets to allow manual OPENING above 83 psig INCREASING.

RefMaterial

QuestionComment D - correct - IAS and SAS systems are cross connected through valves SA-V92 and - V93 which automatically close when service air pressure lowers to 90psig (reopens on 93psig increasing).

Distract1Comment

Distract2Comment

Distract3Comment

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..079.A2.01			079	A2	01
KaRevision				**CognitiveLevel**	**ExamLevel**
				1	R

Examination Outline Cross-reference:

(Page 2) 295003 Partial or Complete Loss of AC G2.1.7
Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.

Level	RO
Tier #	1
Group #	1
K/A #	295003 G2.1.7
Importance Rating	3.7

Proposed Question: **# 14**

From a normal full power operating condition, a complete and instantaneous loss of bus 10500 occurs. All equipment functions as designed. Several minutes after the loss, the bus is still de-energized.

Which of the following is the probable cause?

a. Ground fault trip of circuit breaker 10514
b. Loss of DC Control Power to the 10500 Bus
c. Overcurrent condition on CRD Pump "Alpha" Motor
d. Actuation of the 10500 Degraded Bus Voltage Timer

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – Per ARP-09-8-2-8, a trip of 10514 (10300 – 10500 Tie Breaker) is a cause of the loss of 10500. The ground fault trip is an "86" Device trip which requires a manual reset.

b. Incorrect – Loss of DC Control Power to the 10500 breaker(s) would not generate a trip (result in loss) but would result in a loss of breaker position indications.

c. Incorrect – Via selective tripping methodology, the CRD Pump breaker, alone, should trip on overcurrent. If that didn't work, then L15 power supply breaker should trip. It would take failure of both selective trips to propagate to the entire 10500 bus, and credibility of occurrence would be a stretch at best.

d. Incorrect – The "several minutes" portion of the stem, along with the absence of any delineated EDG failure(s), nullifies distracter "d" as this would trip the breaker momentarily but only until EDGs came up to speed and closed back in.

Technical Reference(s): AOP-18 (Attach if not previously provided)

ARP-09-8-2-8

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E, EO 1.05.c & 1.10 (As available)

SDLP-71O, EO 1.23

Question Source: Bank # 25599

Modified Bank # New

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 4) 295024 High Drywell Pressure / 5 EK3.05

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE:

- RPV flooding

Level	RO
Tier #	1
Group #	1
K/A #	295024 EK3.05
Importance Rating	3.5

Proposed Question: **# 15**

Which of the below scenarios describes a condition which may require entry into EOP-7, RPV Flooding, and the reason for that entry?

a. Several minutes after a LOCA, Drywell Pressure is 5 psig and Drywell Temperature is 240 degrees with no RHR pumps available. EOP-7 entry enables flooding the RPV to the level of the pipe break thus effecting Drywell Spray through the pipe break.

b. RPV Water level below the Top of Active Fuel with RPV Pressure at 400 psig and only SLC injecting to the RPV. EOP-7 entry will perform the Emergency Depressurization that enables the low pressure injection sources to inject to the RPV.

c. **Several minutes after an unisolable large break LOCA, Drywell Pressure is 25 psig and all Drywell Temperatures are above 320 degrees. EOP-7 entry enables the use of all available injection sources to raise RPV water level to the Main Steam Lines.**

d. RPV Water Level on the Fuel Zone Recorder and Refuel Zone is 170 inches with Drywell Temperatures 150 to 200 degrees. All other RPV Level Indicators are unavailable. EOP-7 will restore RPV Water Level to the usable range of the Refuel Zone Level Indicator.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: The conditions presented describe a DBA-LOCA which results in pressure equalization between the RPV and the Drywell. Using the Drywell Pressure value on the EOP-11 RPV Saturation curve, you find yourself on the bad side indicating that level instrumentation MAY be unreliable due to boiling in the reference leg. Should this occur, a conclusion that RPV level cannot be determined will lead to direction to enter EOP-7.

a. Incorrect: These parameters are on the bad side of the Drywell Spray Initiation Pressure Limit and the EOP-4 Drywell Spray Leg is not used until Torus Pressure exceeds 15 psig. Without additional information the candidate is forced to conclude that Torus Pressure is approximately 3 psig.

b. Incorrect: Although EOP-7 entry will accomplish the ED enabling Low Pressure injection, the conditions described do not result in EOP-7 entry. Per the Alternate RPV Level Control Leg of EOP-2, these conditions result in waiting until RPV level drops to -19 inches before an ED is required.

d. Incorrect: Although the conditions describe a significant degradation in RPV Level Indication, they do not describe the inability to determine level. The correct course of action would be to raise RPV level 5 inches to the useable range.

Technical Reference(s): EOP-2,4,7,and 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP-2, 4, 7 and 11

Learning Objective: MIT-301.11.B EO 1.01, Mit-301.11.C EO-1.03, 1.05, and 1.07

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 23) G2.3.9
Knowledge of the process for performing a containment purge.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	Cat 3 G2.3.9	
Importance Rating	2.5	

Proposed Question: **# 16**

The plant is conducting a shutdown, power is currently 30% and lowering. It is desired to de-inert the Primary Containment (both the Drywell and Torus) as soon as possible to permit containment access for maintenance.

Which procedurally allowed flowpath would be used for de-inerting the Primary Containment?

a. Through the Standby Gas Treatment System with the Drywell and Torus de-inerted simultaneously.

b. **Through the Standby Gas Treatment System with the Drywell de-inerted first and then the Torus de-inerted.**

c. Through the Reactor Building Ventilation System with the Drywell and Torus de-inerted simultaneously.

d. Through the Reactor Building Ventilation System with the Drywell de-inerted first and then the Torus de-inerted.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT – Test of OP-37, Section F, Caution on simultaneous opening of Drywell and Torus Exhaust Isolation Valves bypassing the pressure suppression function of the Torus when Primary Containment is required. Per stem power conditions, Primary Containment is required

a. Incorrect – See above. Word "simultaneously" makes incorrect.

c. Incorrect – See above. Wrong system and "simultaneously."

d. Incorrect – See above. Wrong system with correct order.

Technical Reference(s): OP-37, Section F (Attach if not previously provided)

TS 3.6.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16B, EO 1.13.c/d (As available)

SDLP-01B, EO 1.06.b

Question Source: Bank # 25686

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 18) 202002 Recirculation Flow Control K5.02

Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM:

- Feedback signals

Level	RO
Tier #	2
Group #	2
K/A #	202002 K5.02
Importance Rating	2.6

Proposed Question: **# 17**

The plant is operating at normal 100% power, steady-state, when Alarm 09-4-3-40, "RWR MG B SPEED CNTRL SIG FAILED" is received. All plant operating parameters are within required bands at the time of the alarm.

Which of the following is expected to occur as a result of this condition?

a. Alarm 09-4-3-20 "RWR MG B SCOOP TUBE LOCK" only

b. Alarm 09-4-3-39 "RWR MG B 30% OR 44% SPEED LIMIT" only

c. Alarm 09-4-3-10 "RWR MG B GEN LOCKOUT" concurrent with Alarm 09-4-3-19 "RWR MG B DRV MTR BKR TRIP"

d. Alarm 09-4-3-39 "RWR MG B 30% OR 44% SPEED LIMIT" concurrent with the "RWR MG B RUNBACK" red light on the 09-4 apron

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Per ARP 09-4-3-40, a loss of signal to the scoop tube positioner results in an automatic scoop tube lock; which would yield the subsequent 09-4-3-20 alarm. ARP 09-4-3-20 validates cause/effect.

b. Incorrect: Per ARP 09-4-3-40, a loss of signal to the scoop tube positioner results in an automatic scoop tube lock **only**; which would **not** yield the subsequent 09-4-3-39 alarm only (indicative of 30% runback) without plant parameters warranting the runback (stem parameters normal).

c. Incorrect: Per ARP 09-4-3-40, a loss of signal to the scoop tube positioner results in an automatic scoop tube lock **only**; which would **not** yield the subsequent 09-4-3-10 / 09-4-3-19 alarms without some type of concurrent overvoltage / overcurrent condition, which the stem lacks.

d. Incorrect: Per ARP 09-4-3-40, a loss of signal to the scoop tube positioner results in an automatic scoop tube lock **only**; which would **not** yield the subsequent 09-4-3-39 alarm / apron red light (indicative of 44% runback) without the scoop tube being in Auto-Unlock and plant parameters warranting the runback (stem parameters normal).

Technical Reference(s): ARP 09-4-3-39 (Attach if not previously provided)

ARP 09-4-3-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I, EO 1.05.a.8/b.5/c.3, 1.12.b, & 1.14.d (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 23) G2.3.4
Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized.

Level	RO
Tier #	3
Group #	[redacted]
K/A #	Cat 3 G2.3.4
Importance Rating	2.5

Proposed Question: **# 18**

Which of the below individuals has the authority to approve radiation exposures in excess of 10CFR20 limits?

a. Emergency Director
b. Radiation Protection Manager
c. Vice President of Plant Operations
d. Technical Support Center (TSC) Manager

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – Per EAP-15, ONLY the Emergency Director can authorize exposure limits in excess of 10CFR20 limits.

b. Incorrect – See above. Can authorize exceeding administrative limits, but not federal limits.

c. Incorrect – See above. Can authorize exceeding administrative limits, but not federal limits.

d. Incorrect – See above. No reference to authorization responsibilities. Put in as another ERO position.

Technical Reference(s): AP-07.05 (Attach if not previously provided)

EAP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.3, EO 1.20.B (As available)

Question Source: Bank # 25683

[redacted] (Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments: This question was SRO-level on the last exam and is being used as an RO-level question on this exam.

Examination Outline Cross-reference:

(Page 20) 271000 Offgas A2.10

Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Offgas system high flow

Level	RO
Tier #	2
Group #	1
K/A #	271000 A2.10
Importance Rating	3.1

Proposed Question: **# 19**

The plant is operating at 100% power when annunciator 09-6-1-29 "CNDSR VAC LO" alarms. Condenser vacuum has slowly dropped to 25.5 inches of Hg and generator output has dropped by 3 MWe. A NPO, dispatched to search for potential causes, has just reported a small tear in the expansion boot between the Main Condenser and the LP turbine hood. How has the Offgas flow changed (prior to any operator action) and what actions will the operator take in response to this event?

a. Offgas flow has RISEN; place the spare Steam Jet Air Ejector(s) in service.
b. Offgas flow has DROPPED; trip Hydrogen Addition and start the Condenser Air Removal Pumps.
c. Offgas flow has RISEN; trip Hydrogen Addition and start the Condenser Air Removal Pumps.
d. Offgas flow has DROPPED; trip the Turbine, Scram the Reactor and close the Main Steam Line Isolation Valves.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – AOP-31 indicates that air in-leakage will be indicated by a rise in Offgas flow rate; and directs placing the spare SJAEs in service.

b. Incorrect – Offgas flow will rise (see above) and Condenser Air Removal Pumps discharge to the 1.75 minute holdup pipe, which is not designed for explosion pressure. For this reason, their operation is not permitted if reactor power is greater than 5%. Power is greater than 5%. Hydrogen Addition trip ok.

c. Incorrect – Offgas flow rise correct, but Condenser Air Removal Pump operation is not permitted if reactor power is greater than 5% (see above). Hydrogen Addition trip ok.

d. Incorrect – Offgas flow will rise (see above) and the combination of vacuum and MWe lowering would not warrant these actions.

Technical Reference(s): AOP-31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-38, EO 1.10.h & 1.14.b (As available)

Question Source: Bank # 21356

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-01-01 (11/5/01)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 21) 286000 Fire K4.01

Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following:

- Adequate supply of water for the fire protection system

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	286000 K4.01	
Importance Rating	3.4	

Proposed Question: **# 20**

A power loss has occurred, rendering 76P-3 (Makeup "Jockey" Fire Pump) inoperable. The Fire Protection System functions as designed. For these conditions, the Electric Fire Pump (76P-2) will automatically start if the fire protection header pressure drops to:

a. 92 psig
b. 95 psig
c. 105 psig
d. 120 psig

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Per OP-33, Electric Fire Pump 76P-2 auto-start setpoint is 105 psig decreasing.

a. Incorrect: Per OP-33, **East Diesel** Fire Pump 76P-**4** auto-start setpoint is 92 psig decreasing.

b. Incorrect: Per OP-33, **West Diesel** Fire Pump 76P-**1** auto-start setpoint is 95 psig decreasing.

d. Incorrect: Arbitrary pressure slightly lower than the normal operating header pressure range of 140 – 160 psig (from OP-33)

Technical Reference(s): OP-33 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76, EO 1.05.c.1/2 & 1.14.f (As available)

Question Source: [redacted] Bank # [redacted]

Modified Bank # 25838 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
Susquehanna 1	8/8/2003	ILO	25838	BWR

[redacted] The T-20 Transformer developed an internal fault causing a fire and automatic actuation of the Fire Protection Deluge System. The Fire Protection System functions as designed. For these conditions the Diesel Driven Fire Pump will automatically start if the fire protection header pressure drops to:

Answer **85 psig.**

Distractor 1 95 psig

Distractor 2 105 psig

Distractor 3 125 psig

RefMaterial

QuestionComment

Distract1Comment The auto start of the Motor Driven Fire pump

Distract2Comment The auto start of the Jockey Fire Pump

Distract3Comment The auto shutdown of the Jockey Fire Pump

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..600000.2.1.31			600000	2.1	31
KaRevision				CognitiveLevel	ExamLevel
				1	R

Examination Outline Cross-reference:

(Page 23) G2.3.11
Ability to control radiation releases.

Level	RO
Tier #	3
Group #	[redacted]
K/A #	Cat 3 G2.3.11
Importance Rating	2.7

Proposed Question: **# 21**

The plant was operating at 100% power when a large steam leak occurred inside the Reactor Building. Standby Gas Treatment System (SGT) Trains "Alpha" and "Bravo" are operating at rated flows. Secondary Containment differential pressure (to atmospheric) is +1.5" Water Gage (WG). All equipment performed as designed.

Off-Site radioactivity releases are expected to be _____ and can be minimized by _____.

a. ground level releases via Standby Gas Treatment (SGT) only; securing one of the operating SGT trains.

b. elevated releases via SGT and ground level releases via Reactor Building Ventilation; securing Reactor Building Ventilation.

c. ground level releases via SGT and Reactor Building Ventilation; securing one of the operating SGT trains and manually isolating Reactor Building Ventilation.

d. elevated releases via SGT and ground level releases via Reactor Building leakage (exfiltration); lowering the pressure in the secondary containment.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Per EAP-4, only the Stack is an elevated release. Everything else is considered ground level. SGT is discharging to the stack and Rx Building exfiltration is considered a ground level. Lowering pressure in the Rx Bldg will have the impact of lowering exfiltration from the building.

a. Incorrect – See above. SGT is not a ground level. Securing one train of SGT would have the impact of lowering release rates though.

b. Incorrect – See above. Rx Bldg Ventilation would be isolated on stem condition d/p and therefore would be in a recirculate/cool mode. Additionally, securing ventilation would have no impact.

c. Incorrect – See above. SGT would not be a ground level release; and, although securing one train of SGT may lower release rates, Rx Bldg Vent is already isolated on stem condition d/p.

Technical Reference(s): OP-51A (Attach if not previously provided)

EOP-5 / EAP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A, EO 1.09.b (As available)
SDLP-66A, EO 1.05.c
SDLP-01B, EO 1.20

Question Source: Bank # [redacted]

Modified Bank # 25680 (attached) (Note changes or attach parent)

Question History: New [redacted] NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
FitzPatrick 1	7/1/2003	ILO	25680	BWR

The plant was operating at 100% power when a large steam leak occurred inside the Reactor Building. SGT Train "A" and "B" are operating at rated flows. Secondary Containment pressure is +1.5" WG.
Off-Site radioactivity release rates are expected to be

Answer **Elevated releases via SGT and Ground releases via Reactor Building leakage**

Distractor 1 Ground releases via SGT only

Distractor 2 Ground releases via SGT and Reactor Building Ventilation

Distractor 3 Elevated releases via SGT and Ground releases via Reactor Building Ventilation

RefMaterial

QuestionComment

Distract1Comment

Distract2Comment

Distract3Comment

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..290001.K3.01			290001	K3	01
KaRevision				**CognitiveLevel**	**ExamLevel**
				2	R

Examination Outline Cross-reference:

(Page 11) 211000 SLC A3.07

Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including:

- Lights and alarms: Plant-Specific

Level	RO
Tier #	2
Group #	1
K/A #	211000 A3.07
Importance Rating	3.7

Proposed Question: **# 24**

The plant experiences an ATWS during 100% power operation. Reactor power, level, and pressure remain at their previous full-power values. The CRS has directed you to initiate Standby Liquid Control (SLC). You operate the keylock switch to "SYS A" and successfully start 11P-2A ("Alpha" SLC Pump).

Which of the following sets of indications (10-seconds after pump start) supports your making the report that "Alpha SLC pump is running and injecting to the reactor"?

	Squib Valve Continuity (SQUIB VLVS READY) Lights	Alarm 09-3-3-30, "SLC SQUIB VLC CONTINUITY LOSS"	SLC Pump 11P-2A Discharge Pressure
a.	Illuminated (*on*)	Extinguished (*no alarm*)	1045 psig
b.	Extinguished (*off*)	Illuminated (*alarming*)	1045 psig
c.	Illuminated (*on*)	Illuminated (*alarming*)	1280 psig
d.	Extinguished (*off*)	Extinguished (*no alarm*)	1280 psig

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: OP-17 provides a startup validation list for SLC. Squib Valve Ready Lights are "ON" and SLC Pump Discharge Pressure is ≥ reactor pressure, which the stem indicates remained at it's previous full-power value of approximately 1040 psig. ARP 09-3-3-30 does not indicate SLC Pump Start as a possible cause for this alarm; and the alarm is "OFF". (verified in simulator)

b. Incorrect: Per OP-17, Squib Ready Lights are "**ON**" and SLC Pump Discharge Pressure is ≥ reactor pressure, which is still correct in this distracter. ARP 09-3-3-30 does not indicate SLC Pump Start as a possible cause; and the alarm is "**OFF**".

c. Incorrect: Per OP-17, Squib Ready Lights are "ON" and SLC Pump Discharge Pressure is ≥ reactor pressure, which is still correct in this distracter, but would **not** be expected to be this high. ARP 09-3-3-30 does not indicate SLC Pump Start as a possible cause; and the alarm is "**OFF**".

d. Incorrect: Per OP-17, Squib Ready Lights are "**ON**" and SLC Pump Discharge Pressure is ≥ reactor pressure, which is still correct in this distracter, but would **not** be expected to be this high. ARP 09-3-3-30 does not indicate SLC Pump Start as a possible cause; and the alarm is "OFF".

Technical Reference(s): OP-17, Section D (Attach if not previously provided)

GE 791E462

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11, EO 1.05.b.4 & 1.12.a (As available)

Question Source: Bank # [redacted]

Modified Bank # 25852 (attached) (Note changes or attach parent)

Question History: New [redacted] NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
Susquehanna 1	8/8/2003	ILO	25852	BWR

An ATWS transient occurs and boron must be injected with the Standby Liquid Control System (SLC).

Which of the following describes the effect of a successful initiation of SLC on these indications:

1) Squib Valve Ready (Continuity) lights
2) Alarm on Panel 1C601, SBLC SQUIB VALVES LOSS OF CKT CONTINUITY AR-107-A03
3) Pump discharge pressure

Answer

1) Extinguished
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

Distractor 1

1) Illuminated
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

Distractor 2

1) Illuminated
2) Annunciator not in Alarm.
3) Just above reactor pressure

Distractor 3

1) Extinguished
2) Annunciator not in Alarm
3) Just above reactor pressure

RefMaterial

QuestionComment correct answer, Squib valve fires, causing loss of continuity, alarm annunciates indicating loss of continuity, pumps start with discharge pressure slightly greater than reactor pressure.

Distract1Comment Continuity lights go out.

Distract2Comment Continuity lights go out. Alarm is annunciated, discharge pressure 200 psig reactor pressure.

Distract3Comment Alarm is annunciated, discharge pressure 200 psig above reactor pressure.

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..211000.A4.08			211000	A4	08
KaRevision				CognitiveLevel	ExamLevel
				1	R

Examination Outline Cross-reference:

(Page 11) 215003 IRM K6.05

Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM:

- Trip units

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215003 K6.05	
Importance Rating	3.1	

Proposed Question: **# 25**

The Reactor Mode Switch has just been placed in "RUN" during a plant startup. Subsequently, a low detector voltage condition on Intermediate Range Monitor (IRM) "Golf" has brought in Alarm 09-5-2-52 "IRM TRIP SYS A INOP OR UPSCALE TRIP". The result is...

a. alarm only, as all IRM protective actions are bypassed.
b. a control rod block occurs, if APRM "Golf" is also downscale.
c. a half-scram on RPS "Alpha" occurs, if APRM "Echo" is also downscale.
d. an immediate control rod block and a half-scram on RPS "Alpha" occur.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Per ARP 09-5-2-52(53), a low detector voltage is indicated as a cause for the alarm, and automatic actions are dependent on mode switch position. With mode switch in "RUN", the result is a RPS Half-Scram on that side if the companion APRM is downscale. With the mode switch NOT in "RUN", a rod block and a RPS Half-Scram on that side would occur.

a. Incorrect: See above – only rod blocks bypassed in "RUN"

b. Incorrect: See above.

d. Incorrect: See above.

Technical Reference(s): ARP 09-5-2-52(53) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.05.a.4.h, 1.14.f, & 1.15.g (As available)

Question Source: [redacted] Bank # [redacted]

Modified Bank # 25642 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
FitzPatrick 1	7/1/2003	ILO	25642	BWR

An IRM HI Flux Control Rod Block is automatically bypassed when_____?

Answer **The Reactor Mode Switch is placed in RUN.**

Distractor 1 The IRM is on Range 1.

Distractor 2 The IRM's companion APRM is downscale.

Distractor 3 The SRM's are fully inserted.

RefMaterial

QuestionComment

Distract1Comment

Distract2Comment

Distract3Comment

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..215003.A3.04			215003	A3	04
KaRevision				**CognitiveLevel**	**ExamLevel**
				1	R

Examination Outline Cross-reference:

(Page 10) 203000 RHR/LPCI: Injection Mode G2.3.4

Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized.

Level	RO
Tier #	2
Group #	1
K/A #	203000 G2.3.4
Importance Rating	2.5

Proposed Question: **# 26**

You are the SNO during a significant plant transient in which a General Emergency has just been declared. None of the Emergency Response Facilities have reported in as staffed. The only RPV injection source is "A" RHR yet due to a power supply loss, you have been directed to dispatch a NPO to manually open 10MOV-25A. Your RP Tech reports that dose rates in the area of 10MOV-25A are 300 Rem/Hour. Which of the below actions are required to be performed, by your Control Room Supervisor, to complete this task?

a. No additional permissions are required. Dispatch the NPO to the area above the Drywell Personnel Access. Instruct him that he is limited to 5 minutes to complete the task.

b. No additional permissions are required. Dispatch the NPO to the Reactor Building 300 Elevation West. Instruct him that he is limited to 10 minutes to complete the task.

c. Obtain Shift Manager permission. Dispatch the NPO to the area above the Drywell Personnel Access. Instruct him that he is limited to 5 minutes to complete the task.

d. Obtain Shift Manager permission. Dispatch the NPO to the Reactor Building 300 Elevation West. Instruct him that he is limited to 10 minutes to complete the task.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Until the Emergency Director has taken over for the Shift Manager, the SM has all ED responsibilities including the authorization of Emergency Exposures. This responsibility cannot be delegated to the CRS as many SM duties can. This emergency exposure is clearly required to protect the public and therefore is allowed 25 Rem. The location of the 10RHR-25A manual operator is in the LPCI Mezzanine above the Drywell Personnel Access.

a. Incorrect: This exposure requires Shift Manager/Emergency Director permission and this permission cannot be delegated. The time allotment and locations are correct.

b. Incorrect: This exposure requires Shift Manager/Emergency Director permission and this permission cannot be delegated. The time allotment is another Emergency Exposure limitation intended to protect valuable property. The location given is the location of the "A" Core Spray injection valves vice RHR.

d. Incorrect: The permission statement is correct. The location is that of the "A" Core Spray injection valves vice RHR and the time allotment is another Emergency Exposure limitation intended to protect valuable property.

Technical Reference(s): EAP-15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO 1.11.c.3
EP-12.5.4.2, EO 3.06; EP-12.5.3, EO 1.20

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 3) 295016 Control Room Abandonment G2.1.30
Ability to locate and operate components / including local controls.

Level	RO
Tier #	1
Group #	1
K/A #	295016 G2.1.30
Importance Rating	3.9

Proposed Question: **# 27**

The Shift Manager has just ordered a control room evacuation in accordance with AOP-43, "Plant Shutdown From Outside the Control Room." When you arrive at the Auxiliary Shutdown Panel 25 RSP you place the isolation switch for LPCI INBOARD INJECTION VALVE (10MOV-25B) in LOCAL. Reactor water level is 50 inches and reactor pressure is 700 psig. Reactor water level and reactor pressure are dropping. With the isolation switch in LOCAL which statement below describes the operation of the LPCI INBOARD INJECTION VALVE (10MOV-25B) valve as reactor pressure and reactor water level drop.

a. Must be manually opened at valve (10MOV-25B) location
b. Will automatically open when the "Bravo" RHR pump is started
c. Will automatically open when reactor pressure is less than 450 psig
d. Must be manually opened from the Auxiliary Shutdown Panel 25 RSP

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – When the isolation switch for the 10MOV-25B is placed in LOCAL this disables all interlocks and will only allow operation of this valve from the Auxiliary Shutdown Panel. In addition, the valve will not open when the pump is started, the interlock is based on an ECCS signal and pressure.

a. Incorrect – See above

b. Incorrect – See above

c. Incorrect – See above.

Technical Reference(s): AOP-43 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO 1.05.a.8.d & 1.11.b.9/23.a (As available)

Question Source: Bank # 21353

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-01-01 (11/5/01)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 19) 219000 RHR/LPCI: Torus/Pool Cooling Mode K6.06

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI TORUS/SUPPRESSION POOL COOLING MODE:

- Suppression pool

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	219000 K6.06	
Importance Rating	3.7	

Proposed Question: **# 28**

PLACE OP-13A, ATTACHMENT 1 INTO EXAM FORMAT OR PROVIDE AS EXAM ATT.

During accident conditions, the "Alpha" Loop of RHR is operating in Torus cooling with two (2) RHR pumps and the following indications:

- "A" RHR Loop Flow is 10,000 gpm.
- Torus Water Temperature is 193°F.
- Torus Pressure is 0 psig.
- Torus Water Level is 7.0 feet.

Continued operation of the "A" loop of RHR under these conditions....

a. may result in pump damage. The 0 psig overpressure curve is violated by 35 °F.
b. **may result in pump damage. The 0 psig overpressure curve is violated by ~ 10°F.**
c. will not damage the pumps. The 5 psig overpressure curve has a 1° F margin.
d. will not damage the pumps. The 5 psig overpressure curve has an ~ 20°F margin.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: Proper application of the curve will find an operating point at 5000 gpm and 193° torus temperature which is approximately 10° in violation of the 2 pump curve. 10° is chosen to provide validity to candidate curve interpolation at the computed overpressure value of 2 psig.

a. Incorrect: Incorrect application at the 10000 gpm line will find a 0 psig overpressure violation of the 2 pump curve by ~35°.

c. Incorrect: The 5 psig overpressure curve is incorrect but may be used if the candidate does not use the .4 multiplier in determining the value of overpressure. On the 10000 gpm line, the 5 psig overpressure for 2 pump operation is safe by ~1°.

d. Incorrect: The 5 psig overpressure curve is incorrect but may be used if the candidate does not use the .4 multiplier in determining the value of overpressure. On the 5000 gpm line, the 5 psig overpressure for 2 pump operation is safe by 20°.

Technical Reference(s): OP-13A, Attachment 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: OP-13A, Attachment 1

Learning Objective: MIT-301.11B EO-1.01 (As available)

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

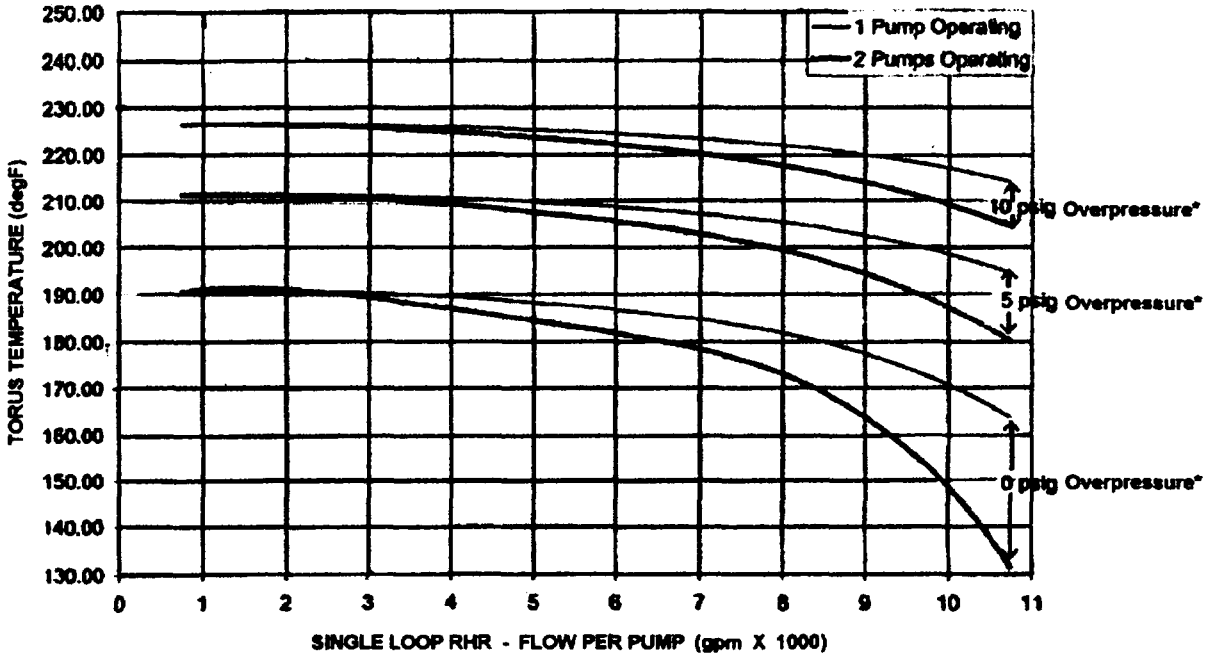
10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

RHR NPSH AND VORTEX LIMITS

RHR PUMP NPSH LIMIT



*Torus Overpressure = Torus Pressure + 0.4 (Torus Water Level - 1.92)

VORTEX LIMIT

Torus Water Level **Greater Than or Equal to** 8.92 feet

Examination Outline Cross-reference:

(Page 10) 209001 LPCS K6.03

Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM:

- Torus/suppression pool water level

Level	RO
Tier #	2
Group #	1
K/A #	209001 K6.03
Importance Rating	3.3

Proposed Question: **# 29**

The Core Spray System Pumps (14P-1A /1B) are normally aligned to take suction on the _____; and during pump operation, if level is continuously lost from the normal suction path, then _____.

a. Torus; the Core Spray Pumps would continue to run until they overheat and trip.
b. Torus; the Core Spray Pump Suction Valve(s) will automatically transfer to the CSTs.
c. CSTs; the Core Spray Pump Suction Valve(s) will automatically transfer to the Torus.
d. CSTs; the Core Spray Pumps will trip on low suction pressure if the suction source is not manually swapped.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Per OP-14 and supported by pump / suction valve circuitry drawings, the CS Pumps do not have Low Suction Pressure trips and there is no auto-swap on the Torus and CST suctions in the case of Core Spray. Core Spray is normally aligned to the Torus, but can be manually aligned to the CSTs (requires declaring respective pump inop). A pump running until it loses suction would be a similar condition to a pump with a sheared shaft and would continue to operate until, in this case, friction builds up to a point where overheating occurs and the rotor may seize on bearing failure for example, which in turn would trip the pump on overcurrent conditions.

b. Incorrect: See above. First part correct, but there is no auto-swap feature on the suction valves.

c. Incorrect: See above. First part incorrect and there is no auto-swap feature on the suction valves.

d. Incorrect: See above. First part incorrect and there is no low suction pressure trip on the Core Spray Pumps.

Technical Reference(s): OP-14 (Attach if not previously provided)

ESK-5BF

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14, EO 1.05.c.1 & 1.10.b (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 11) 212000 RPS K3.12

Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on the following:

- Secondary containment integrity

Level	RO
Tier #	2
Group #	1
K/A #	212000 K3.12
Importance Rating	3.2

Proposed Question: **# 32**

The plant is operating in a normal, full-power lineup when a loss of RPS Bus "Alpha" occurs. In accordance with the Prompt Actions of AOP-59, "Loss of RPS Bus A Power", you are directed by the CRS to isolate Reactor Building Ventilation per Section G of OP-51A, "Reactor Building Ventilation and Cooling System".

As you approach Control Room Panel 09-75 to perform the directed action, you discover which of the following Reactor Building Ventilation lineups?

	Intake / *Exhaust* Isolation Valves (66AOV-100A/B) *(66AOV-101A/B)*	Recirculation Dampers (66AOD-105) (66AOD-108)	Rx Bldg Vent. Supply Fans (66FN-5A/B/C)	Exhaust Fans Above Refuel Floor (66FN-13A/B) / *Below Refuel Floor (66FN-12A/B)*
a.	All Valves Closed	Both Closed	No Fans Running	No Fans Running
b.	100A/101A Failed Open 100B/101B Closed	Both Open	No Fans Running	1 "Above Refuel Floor" Fan Running
c.	100A/101A Closed 100B/101B Open	Both Open	2 of 3 Running	1 "Below Refuel Floor" Fan Running
d.	All Valves Open	Both Closed	2 of 3 Running	1 Fan in Each Set Running

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: OP-51A, Page 20. Action directed by AOP-59 to "isolate" is to complete the isolation. Rx Bldg Ventilation will already be in an "isolate"/recirculation mode as 66AOV-100A/101A will have received a valid signal to "isolate". Remaining damper/fan configurations are those expected for the "isolate" mode.

a. Incorrect: 66AOV-100B/101B would be **Open**. Recirc Dampers would be **Open**. **Two** (2) Supply Fans would be running, versus none; and one "Below Refuel Floor Exhaust Fan, versus **None**, would be running.

b. Incorrect: 66AOV-100A/101A do **not** Fail Open. Recirc Damper position correct. **Two** (2) Supply Fans would be running, versus none; and a "Below Refuel Floor Exhaust Fan, versus **Above** Refuel Floor, would be running.

d. Incorrect: 66AOV-100A/101A would be **Closed**. Recirc Dampers would be **Open**. Supply Fan lineup correct; and **only** one "Below Refuel Floor Exhaust Fan, versus **One in Each Set**, would be running.

Technical Reference(s): OP-51A (Attach if not previously provided)

AOP-59

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05, EO 1.09.a (As available)
SDLP-17, EO 1.05.c.1 & 1.10.e
SDLP-66A, EO 1.05.c.1/2, 1.06.b, & 1.10.d

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 12) 215004 Source Range Monitor A4.05

Ability to manually operate and/or monitor in the control room:

- SRM back panel switches, meters, and indicating lights

Level	RO
Tier #	2
Group #	1
K/A #	215004 A4.05
Importance Rating	3.1

Proposed Question: **# 34**

The following plant conditions exist:

- Reactor Mode Switch is in "STARTUP/HOT STBY"
- All Intermediate Range Monitors (IRM's) are on Range 3 or 4
- Source Range Monitor (SRM) "Alpha" is reading 0.5 cps
- SRMs "Bravo" and "Charlie" are reading 8.3×10^4 cps
- SRM "Delta" mode switch is in "STANDBY"

A rod block signal has been generated. Based on the above indications, which one of the following has caused the rod block?

a. SRM Inoperable
b. SRM Downscale
c. SRM Upscale
d. SRM Not Full-In

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – SRM "D" Mode Switch being placed in "Standby", on back Panel 09-12, will result in a SRM Inop alarm with a concurrent rod block. This function is not bypassed until IRM Range 8 or Reactor Mode Switch in "RUN".

b. Incorrect – Although SRM "A" is downscale, this rod block function is bypassed with associated IRMs on Range 3 or above.

c. Incorrect – Although SRMs "B" and "C" are approaching the Hi Flux (Upscale) Alarm setpoint of 1×10^5 cps, they are not yet there.

d. Incorrect – this rod block function is bypassed with > 100 cps on the respective SRM, associated IRMs on Range 3 or above, or Reactor Mode Switch in "RUN".

Technical Reference(s): ARP 09-5-2-21 / 31 / 51 (Attach if not previously provided)

OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.05.a.5.a, 1.12, 1.14.a/b/c & 1.15.b/c/d (As available)

Question Source: Bank # 25643

Modified Bank # New (Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 22) G2.2.30
Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area / communication with fuel storage facility / systems operated from the control room in support of fueling operations / and supporting instrumentation.

Level	RO	[redacted]
Tier #	3	
Group #	[redacted]	
K/A #	Cat 2 G2.2.30	
Importance Rating	3.5	[redacted]

Proposed Question: **# 35**

The plant is in a refueling outage. The spent fuel pool gates are removed. Control Rod Drive stroke time testing is in progress. Thirty, (30) minutes after a complete loss of Instrument Air occurs, you note (on Panel 09-4) that Refuel Water Level (02-3LI-86) indicates that RPV Level has risen several inches over the last hour.

Which of the below is the probable cause?

a. In-Service Fuel Pool Filter/ Demineralizer has isolated
b. In-Service CRD Flow Control Valve (03FCV-19A/B) failed open
c. RWCU Blowdown Flow Control Valve (12FCV-55) failed closed
d. Feedwater Low Flow Control Valve, (34FCV-137), loss of air signal

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT – Question forces conclusion that CRD is in service and the RWCU Blowdown Mode is being used for level control. With the fail position of 12FCV-55 being closed, RPV Water Level will begin to rise.

a. Incorrect – See above. FPC filter demin status has no effect on level.

b. Incorrect – See above. CRD FCV's fail closed on loss of air yet are pinned to prevent full closure.

d. Incorrect – See above. FW Low Flow Control Valve fails as-is.

Technical Reference(s): AOP-12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39, EO 1.09.f/j (As available)
LPAOP, EO 1.02 & 1.10

Question Source: Bank # 25605
Modified Bank #
New
(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 3) 295018 Partial or Total Loss of CCW AA2.05

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

- System pressure

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295018 AA2.05	
Importance Rating	2.9	

Proposed Question: **# 37**

At 0915, ALARM 09-6-2-31 "RBC HDR PRESS LO" is received in the Control Room. RBC header pressure continues to trend downward at 5 psig/minute. If all components and instrumentation function as designed, which of the following events would be expected to occur at 0922?

a. The standby RBCLC Pump will start automatically and assume load to ensure cooling of all RBCLC Safe-Shutdown components

b. Alarm 09-6-2-32 "RBC HDR PRES LO-LO" will actuate and ESW Lockout Matrix will actuate to inject ESW into the RBCLC System

c. Alarm 09-6-2-22 "RBC MAKEUP TK LVL HI OR LO" will actuate and ESW will automatically align to pick up all RBCLC Safe-Shutdown loads

d. ESW Pump 46-P2A will start automatically and align to pick up the load on the Reactor Water Cleanup (RWCU) Non-Regenerative Heat Exchanger

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: At 0915, the first alarm is received at a system pressure of 75 psig. As pressure trends down at 5 psig/min over 7 minutes, the resulting pressure of 40 psig would bring in the LO-LO alarm and energize the ESW Lockout Matrix to inject ESW to RBCLC.

a. Incorrect: In that the standby RBCLC pump would have started at time 0915 on the first alarm (75 psig), not 40 psig. Additionally, none of the RBCLC loads are required for safe shutdown of the facility.

c. Incorrect: Although the RBCLC Makeup Tank Low Level Alarm could feasibly be received at time 0922, there exists no information to lead the candidate down the path of deducing a linear relationship between a leak size and a pressure/level decrease. Additionally, ESW does <u>not</u> automatically align to supply <u>any</u> RBCLC load; as none are required for safe shutdown of the facility.

d. Incorrect: Although ESW Pump 46-P2A will have started (both ESW Pumps start on Lockout Matrix), ESW cannot be aligned to the RWCU NRHX, manually or automatically, and it will isolate on high outlet temperature.

Technical Reference(s): ARP 09-6-2-31/32 (Attach if not previously provided)

AOP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15, EO 1.05.a.1/c.5, 1.06.b, 1.09.d, & 1.14.b/c (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 3) 295006 SCRAM AK2.01

Knowledge of the interrelations between SCRAM and the following:

- RPS

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295006 AK2.01	
Importance Rating	4.3	

Proposed Question: **# 38**

The reactor is operating at 25% power with recirculation flow at minimum.

If a turbine trip occurs and the bypass valves fail to open, which of the following would be the appropriate procedure(s) to respond to the event?

a. AOP-2, "Main Turbine Trip Without Scram" and AOP-6, "Malfunction of EHC Pressure Regulator"

b. AOP-1, "Reactor Scram" and AOP-6, "Malfunction of EHC Pressure Regulator"

c. AOP-2, "Main Turbine Trip Without Scram" and EOP-2, "RPV Control"

d. AOP-1, "Reactor Scram" and EOP-2, "RPV Control"

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Conditions would cause a high pressure Scram and entry into EOP-2 on high pressure and/or < 177-inches reactor water level.

a. Incorrect – Due to initial power level of 25%, if Bypass Valves were to function correctly, AOP-2 would be entered. As a result of the failure of Bypass Valves, the resultant high pressure condition will Scram the plant. There is no stem indication of an EHC malfunction.

b. Incorrect – Conditions would cause a high pressure Scram, and thus entry into AOP-1. But, there is no stem indication of an EHC Pressure Regulator malfunction; and thus, no entry into AOP-6.

c. Incorrect – Following the Scram, EOP-2 would be entered on high pressure and/or < 177-inches reactor water level. As discussed above, AOP-2 would have been entered if Bypass Valves had functioned.

Technical Reference(s): AOP-1 (Attach if not previously provided)
EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05, EO 1.07.a.10 (As available)
SDLP-94A, EO 1.09.f

Question Source: Bank # 25654
Modified Bank # [redacted]
New [redacted]
(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑
55.43 ☐

Comments: *This was an SRO Only question on the last NRC Exam, but is being utilized as an RO Question on this exam.*

Examination Outline Cross-reference:

(Page 3) 295021 Loss of Shutdown Cooling AA1.06

Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING:

- Containment / drywell temperature

Level	RO
Tier #	1
Group #	1
K/A #	295021 AA1.06
Importance Rating	2.8

Proposed Question: **# 39**

The plant was scrammed 8 hours ago and is currently in a normal shutdown cooling lineup. The Drywell is still inerted and over a period of 30 minutes, the following indications are observed:

- Reactor coolant temperature is trending upward
- RPV metal temperatures are trending upward
- Reactor Water Level, as read on the Refueling Level Instrument, is trending upward
- Reactor Water Level, as read on the Fuel Zone Level Instrument, trended slightly downward and then appears to have stabilized at approximately 2 inches below the previous value

Which of the following events has occurred and what is the cause of the Reactor Water Level indication mismatch?

a. A Recirculation Pump was started and flow through the Jet Pumps has introduced error into the Reactor Water Level instrumentation.

b. A Loss of Shutdown Cooling has occurred and a rising Drywell temperature has introduced error into the Reactor Water Level instrumentation.

c. A Recirculation Pump was secured and Natural Circulation through the core has introduced error into the Reactor Water Level instrumentation.

d. A Loss of Shutdown Cooling has occurred and the resulting core Natural Circulation has introduced error into the Reactor Water Level instrumentation.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: The first two indications are primary symptoms of a Loss of Shutdown Cooling, as detailed in AOP-30. IF shutdown cooling is lost, Drywell Temperature will trend upward steadily based on core decay heat one day after shutdown. With a rising Drywell Temperature, as the result of having the largest reference leg vertical drop in the Drywell, the Refueling Level Instrument will experience the worst case rise in indicated level. The Fuel Zone instrumentation has a very small reference leg drop and a slight reduction in indicated level will be introduced by a rising Drywell Temperature.

a. Incorrect: While Jet Pump flows will introduce an error in the Fuel Zone instrumentation, it causes a rise in indicated level versus the slight lowering as indicated above in the stem indications.

c. Incorrect: While natural circulation through the core will introduce an error in the Fuel Zone instrumentation, it also causes a rise in indicated level versus the slight lowering as indicated above in the stem indications.

d. Incorrect: Different precursor leading to the same result of natural circulation. While natural circulation through the core will introduce an error in the Fuel Zone instrumentation, it also causes a rise in indicated level versus the slight lowering as indicated above in the stem indications.

Technical Reference(s): AOP-30 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02B, EO 1.10.e (As available)
SDLP-10, EO 1.09.c/e/g

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 13) 217000 RCIC K6.04

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC):

- Condensate storage and transfer system

Level	RO
Tier #	2
Group #	1
K/A #	217000 K6.04
Importance Rating	3.5

Proposed Question: **# 41**

The loss of Condensate Storage Tank (CST) level can impact Reactor Core Isolation Cooling in two distinctly different ways. First, with RCIC in standby, the loss of CST level could result in _____; and second, during RCIC operation, a lowering of CST level _____.

a. severe water hammer when RCIC injection is initiated; to 59.5 inches will result in annunciator 09-6-2-10 "CST A OR B LVL HI OR LO" and a RCIC Turbine Trip on low suction pressure.

b. **severe water hammer when RCIC injection is initiated; to 59.5 inches will result in annunciator 09-4-0-28 "RCIC CST A LVL LO" and automatic closure of 13MOV-18 (RCIC Pump CST Suction Isolation Valve).**

c. having to manually swap RCIC suction over to the Torus; to 138.9 inches will result in annunciator 09-6-2-10 "CST A OR B LVL HI OR LO" and having to manually isolate all non-ECCS suctions from the CST.

d. having to manually swap RCIC suction over to the Torus; to 138.9 inches will result in annunciator 09-4-0-28 "RCIC CST A LVL LO" and automatic transfer of RCIC suction from the CSTs to the Torus.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: RCIC normally aligned to CSTs in standby. Height of water in CSTs is sufficient to maintain RCIC piping full of water up to first isolation valve – in effect, a "keep-full" system which prevents water hammer on RCIC startup. Per ARP 09-4-0-28 (29), if CSTs drop to 59.5 inches, auto swap of RCIC suction to Torus occurs, which yields opening of Torus Suction Isolation Valves and Closure of 13MOV-18.

a. Incorrect: See above (first part correct). Wrong value for alarm and auto-swap feature would prevent low suction pressure RCIC Turbine Trip as it is "make before break."

c. Incorrect: Per OP-19, auto-swap will occur whether RCIC in operation or not. Per ARP 09-6-2-10, alarm actually at 238.9 inches, and this alarm has no impact on RCIC operation / alignment whatsoever.

d. Incorrect: See above. Also, wrong CST level value corresponding to alarm.

Technical Reference(s): OP-19 / OP-5 (Attach if not previously provided)

ARPs 09-4-0-28(29) / 09-6-2-10

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO 1.05.a.13/b.3, 1.06.d, & 1.10.f (As available)

SDLP-33, EO 1.05.a.1.a & 1.09.c

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 18) 201002 RMCS A1.01

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including:

- CRD drive water flow

Level	RO
Tier #	2
Group #	2
K/A #	201002 A1.01
Importance Rating	2.8

Proposed Question: **# 42**

A control rod is being continuously withdrawn from position 36 to position 48 during a reactor startup. During withdrawal, Control Rod Drive (CRD) water flow should be approximately _____ and during performance of the subsequent control rod coupling check, CRD water flow should _____.

a. 2 gpm; stabilize at a value lower than 2 gpm.
b. 2 gpm; stabilize at a value higher than 2 gpm.
c. 4 gpm; stabilize at a value lower than 4 gpm.
d. 4 gpm; stabilize at a value higher than 4 gpm.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: OP-25, "CRD Hydraulic System" indicates a flow rate of approximately 2 gpm is required to withdraw a control rod, versus a flow rate requirement of 4 gpm to insert a control rod. With no noted seal deficiencies, etc., stall flows achieved during the rod coupling check will be lower than drive flows.

b. Incorrect: 2 gpm correct (see above). CRD Water **pressure** lowers during the withdraw sequence and would rise at stall flows. If candidate makes an erroneous direct correlation between pressure and flow, the higher value will be chosen.

c. Incorrect: 4 gpm is for **insert** sequence, versus withdraw. The secondary part of the answer, in theory, is correct.

d. Incorrect: 4 gpm is for **insert** sequence, versus withdraw. CRD Water **pressure** lowers during the withdraw sequence and would rise at stall flows. If candidate makes an erroneous direct correlation between pressure and flow, the higher value will be chosen.

Technical Reference(s): OP-25 (Attach if not previously provided)

ST-23B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C, EO 1.05.a.6/c.4, 1.08.d/j, & 1.13.k (As available)

SDLP-03F, EO 1.08.c & 1.13.d

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 10) 206000 HPCI A1.07

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including:

- System discharge pressure: BWR-2,3,4

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	206000 A1.07	
Importance Rating	3.7	

Proposed Question: **# 43**

To support Post-Maintenance Testing, HPCI has been manually started in the Full Flow Test (CST-CST) mode with HPCI flow control in automatic. The system is stable at 3500 rpm and 2000 gpm.

To establish the prescribed pressure band, the operator throttles open 23MOV-21, TEST VLV TO CST.

Which of the following CORRECTLY describes the response of HPCI Discharge Pressure and TURBINE GOVERNOR CONTROL VALVE (HOV-2) position to this valve manipulation?

a. HPCI Discharge Pressure lowers (decreases)
   HOV-2 Position - Throttles CLOSED
b. HPCI Discharge Pressure rises (increases)
   HOV-2 Position - Throttles OPEN
c. HPCI Discharge Pressure lowers (decreases)
   HOV-2 Position - Remains the Same
d. HPCI Discharge Pressure rises (increases)
   HOV-2 Position - Remains the Same

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – Per Fundamental Pump Laws, if you throttle open on a control valve in the discharge line of a centrifugal pump, discharge pressure (pump head) will decrease and flowrate will increase. With flow control in automatic, the sensed speed/flowrate increase will result in HOV-2 throttling closed to restore the dialed-in, desired flowrate.

b. Incorrect – See above. Discharge Pressure **decreases** and HOV-2 will throttle **closed** in response to an increasing flowrate.

c. Incorrect – See above. HOV-2 **will** change position.

d. Incorrect – See above. Discharge Pressure **decreases** and HOV-2 **will** change position.

Technical Reference(s): OP-15 (Attach if not previously provided)

ST-4N

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23, EO 1.05.a.23.e (As available)

JLP-LOI-Comp02, EO 1.12

Question Source: [redacted] Bank #

Modified Bank # 681 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Plant Duane Arnold 1 | *Question Id* 681
Exam Date 5/25/1999 | *Exam Type* ILO

The plant is at power and operators are performing post maintenance testing on the HPCI turbine in order to exit the HPCI LCO.

HPCI has been manually started in the CST-CST mode using the TEST POT for control. The system is stable at 3500 rpm and 2000 gpm.

At this point, the operator throttles open CV-2315, TEST BY-PASS, to achieve 3000 GPM.

Which of the following CORRECTLY describes the response of HPCI Discharge Pressure and HV-2200, TURBINE CONTROL VALVE, position to this valve manipulation?

Answer **HPCI Discharge Pressure DECREASES**
HV-2200 position REMAINS THE SAME

Distractor 1 HPCI Discharge Pressure INCREASES
HV-2200 position REMAINS THE SAME

Distractor 2 HPCI Discharge Pressure INCREASES
HV-2200 position THROTTLES OPEN

Distractor 3 HPCI Discharge Pressure DECREASES
HV-2200 position THROTTLES OPEN

RefMaterial

QuestionComment RO must know that the test pot controls HPCI speed, not flow, so HV-2200 will not change position. Pump laws dictate that a pump at constant speed will have a decreased discharge pressure at higher flow.

Distract1Comment Discharge pressure will decrease.

Distract2Comment Discharge pressure will decrease. HV-2200 will not change position.

Distract3Comment HV-2200 will not change position.

CognitiveLevel	ExamLevel	KaNumber
3	R	206000A1.07

KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5	KaRevision
		206000	A1	07	

Examination Outline Cross-reference:

(Page 11) 215003 IRM K5.01

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM:

- Detector operation

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

Proposed Question: **# 44**

A reactor startup is in progress. You are in the process of ranging up on all IRMs. You inadvertently range down on IRM "Charlie", which was indicating a value of 10 on Range 9.

Which of the following will result from the erroneous switch manipulation?

a. IRM "C" reading within OP-65, Startup and Shutdown Procedure" range limits without resulting in a half-scram or rod block.

b. IRM "C" reading outside OP-65, Startup and Shutdown Procedure" range limits without resulting in a half-scram or rod block.

c. IRM "C" reading within OP-65, Startup and Shutdown Procedure" range limits which results in a rod block but no half-scram.

d. IRM "C" reading outside OP-65, Startup and Shutdown Procedure" range limits which results in a half-scram and a rod block.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: OP-65 specifies that IRMs should be changed to maintain between 5 and 75 on the 125 scale. Ranging down from a value of 10 on Range 9 would result in a value of 100 on Range 8. This value would be outside the specified range limits of OP-65, but would not yet have attained the values corresponding to the half-scram or rod block setpoints (120 and 108 respectively)

a. Incorrect: OP-65 specifies IRM range between 5 and 75 on the 125 scale. Ranging down from a value of 10 on Range 9 would result in a value of 100 on Range 8. This value would be **outside** the specified range limits of OP-65, but would not yet have attained the values corresponding to the half-scram or rod block setpoints (120 and 108 respectively)

c. Incorrect: Ranging down from a value of 10 on Range 9 would result in a value of 100 on Range 8. This value would be outside the specified range limits of OP-65, but would not yet have attained the values corresponding to the half-scram or rod block setpoints (120 and 108 respectively)

d. Incorrect: Ranging down from a value of 10 on Range 9 would result in a value of 100 on Range 8. This value would be outside the specified range limits of OP-65, but would not yet have attained the values corresponding to the half-scram or rod block setpoints (120 and 108 respectively)

Technical Reference(s): OP-65 (Attach if not previously provided)

SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07D, EO 1.05.a.4.f & 1.07.d (As available)

LP-OP-65, EO 1.12

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

New X

Question History: Last NRC Exam ____

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 2) 295005 Main Turbine Generator Trip AK1.03

Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP:

- Pressure effects on reactor level

Level	RO
Tier #	1
Group #	1
K/A #	295005 AK1.03
Importance Rating	3.5

Proposed Question: **# 45**

When comparing the various transient analyses contained in the Updated Facility Safety Analysis Report (UFSAR), the _____ transient is the most likely to limit plant operations. This is due to the resulting _____.

a. **Inadvertent High Pressure Coolant Injection Startup at 100% Power**; excess of coolant inventory which may result in severe damage from substantial carryover.

b. **Recirculation Flow Master Controller Failure (Maximum Speed Demand)**; excess of coolant inventory which may result in severe damage from substantial carryover.

c. **Turbine Trip at 100% Power With Failure of the Bypass Valves to Operate;** collapse of voids in the moderator which may result in exceeding fuel cladding safety limits.

d. **Electro-Hydraulic Control (EHC) Pressure Regulator Failure (Rising RPV Pressure)**; collapse of voids in the moderator which may result in exceeding fuel cladding safety limits.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: The only transients discussed as "limiting" in the UFSAR are defined as such due to potential impacts on MCPR. Turbine Trip Without Bypass results in a significant "pressure increase" and analyzing it in that category invokes the remainder of the answer (detailed in Ch 14.5).

a. Incorrect: Although a UFSAR "limiting" transient, it is not a transient that is analyzed in the category of "excess coolant inventory", but is analyzed as "coolant temperature decrease". Thus, carryover is not the concern; increasing density adding positive reactivity is the challenge to clad integrity.

b. Incorrect: Not a "limiting" transient. This is not a transient that is analyzed in the category of "excess coolant inventory", but is analyzed as "coolant flow increase". Thus, carryover is not the concern; reduction of void content and positive reactivity addition is the challenge to clad integrity.

d. Incorrect: Not a "limiting" transient and AOP-6 indicates that this failure will raise RPV Pressure by only approximately 3 psi. In the UFSAR analyses, this transient is considered "mild" with no significant thermal margin reductions.

Technical Reference(s): UFSAR, Chapter 14.5 (Attach if not previously provided)

AOP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: NET-238.07, EO 1.01.b (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 4) 295025 High Reactor Pressure/3 G2.2.12
Knowledge of surveillance procedures.

Level	RO
Tier #	1
Group #	1
K/A #	295025 G2.2.12
Importance Rating	3.0

Proposed Question: **# 46**

During performance of Surveillance Test ST-1I, "Main Steam Isolation Valves Limit Switch Channel Functional Test", if the slow close pushbutton on MSIV 29AOV-86C is depressed too long, ...

a. the MSIV could fully close resulting in a half-scram signal generated from RPS Subchannel "Alpha" One (A1).

b. the MSIV could fully close due to steam forces acting on the valve, resulting in a reactor scram on high reactor pressure.

c. annunciator 09-5-1-2, "MSIVs NOT FULL OPEN TRIP" will alarm and a reactor scram on MSIV closure will occur if > 29% reactor power.

d. annunciator 09-5-1-2, "MSIVs NOT FULL OPEN TRIP" will alarm and a half-scram will be generated from RPS Subchannel "Bravo" Two (B2).

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: Test of Caution in ST-1I in addition to a cause-effect relationship. Caution itself states that if pushbutton held too long, that the steam forces acting on the MSIV could cause it to go full-closed. UFSAR Ch. 14 Transient analyses discuss the impact of closure of one up to and including all MSIV's in the section on transients that cause reactor pressure increases.

a. Incorrect: Although the first portion of the answer is true, the Subchannel A1 reference is incorrect. 29AOV-86C Limit Switches provide input to **A2** and **B1** RPS Trip Logic circuits.

c. Incorrect: Although first portion of the answer is true, the associated RPS logic will allow the full closure of one MSIV without a full scram. In fact, the ST criteria for success, after pulling fuses for the other input to a particular subchannel, is a half-scram. A full scram would not occur on single MSIV position alone.

d. Incorrect: Although the first portion of the answer is true, the Subchannel B2 reference is incorrect. 29AOV-86C Limit Switches provide input to **A2** and **B1** RPS Trip Logic circuits.

Technical Reference(s): ST-1I (Attach if not previously provided)

ARPs 09-5-1-2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29, EO 1.05.b.1, 1.09.f, & 1.13.c (As available)

NET-238.07, EO1.01.c

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 10) 206000 HPCI K5.05

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM:

- Turbine speed control: BWR-2,3,4

Level	RO
Tier #	2
Group #	1
K/A #	206000 K5.05
Importance Rating	3.3

Proposed Question: **# 48**

Operating the High Pressure Coolant Injection (HPCI) Turbine at less than _____ could potentially cause improper oil system operation, overheating of the shaft driven oil pump, and/or _____.

a. 1800 rpm; insufficient exhaust flow resulting in check valve banging
b. 1800 rpm; insufficient Net Positive Suction Head resulting in cavitation
c. 2100 rpm; insufficient exhaust flow resulting in check valve banging
d. 2100 rpm; insufficient Net Positive Suction Head resulting in cavitation

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: OP-15, procedural precaution C.2.3

a. Incorrect: Incorrect rpm but related to other turbines, correct operational implication

b. Incorrect: Incorrect rpm but related to other turbines, incorrect operational implication

d. Incorrect: Correct rpm, incorrect operational implication

Technical Reference(s): OP-15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23, EO 1.13.a (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 12) 215005 APRM / LPRM A3.05

Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including:

- Flow converter/comparator alarms

Level	RO
Tier #	2
Group #	1
K/A #	215005 A3.05
Importance Rating	3.3

Proposed Question: **# 49**

Given the following initial conditions:

- Recirculation Flow Unit "Alpha" has a signal output equal to 92% flow
- Recirculation Flow Unit "Bravo" has a signal output equal to 93% flow
- Recirculation Flow Unit "Charlie" has a signal output equal to 94% flow
- Recirculation Flow Unit "Delta" has a signal output equal to 95% flow

THEN, Flow Unit "Alpha" fails UPSCALE and annunciator 09-5-2-25 "FLOW REF OFF NORM" alarms.

As a result of this condition, APRM _____ will have a _____ trip setpoint (***as compared to the initial setpoint***).

a. Alpha ; Lower

b. Bravo ; Lower

c. Charlie ; Higher

d. Delta ; Higher

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Recirc Flow Units provide a reference core flow signal for APRM and RBM flow-biased trip setpoints via Low-Value Gates (LVG). A failure of Flow Unit "A" upscale will default to the other associated Flow Unit "C" via the auctioneering circuit. APRMs A, C, and E are supplied from Flow Units A & C. Because the Flow Unit "C" signal is higher than the initial Flow Unit "A" signal, the trip setpoints for APRMs A, C, and E will all **rise**.

a. Incorrect: See above.

b. Incorrect: APRMs, B, D, and F are supplied through LVG from Flow Units B & D, and thus would be unaffected.

d. Incorrect: APRMs, B, D, and F are supplied through LVG from Flow Units B & D, and thus would be unaffected.

Technical Reference(s): ARP-09-5-2-25 (Attach if not previously provided)

OP-16 / SDLP-07C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C, EO 1.09.a (As available)

Question Source: [redacted] Bank # [redacted]

Modified Bank # JEX-LOI-SYST02 #48 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

TOPIC: Power Range Monitoring (PRM)

Q TION # 48

Given the following initial conditions:

- Flow Unit A has a signal output equal to 95% flow
- Flow Unit B has a signal output equal to 94% flow
- Flow Unit C has a signal output equal to 93% flow
- Flow Unit D has a signal output equal to 92% flow

THEN Flow Unit D fails UPSCALE. All other Flow Unit signals remain unchanged.

As a result of this condition, APRM ___ will have a _____ trip setpoint.

A. A, Higher
B. B, Higher
C. C, Lower
D. D, Lower

ANSWER: B

LESSON PLAN:	SDLP-07C
OBJECTIVE:	EO-1.09.a
KA:	215005 K3.01 {4.0 / 4.0}
REFERENCES:	SDLP-07C
STION TYPE:	C1
COGNITIVE LEVEL:	Comprehension
AUTHOR:	J Rivers Walsh

JUSTIFICATION:
Recirculation Flow Units provide a reference core flow signal for APRM and RBM flow-biased trip setpoints via low value gates (LVGs). A failure of a flow unit downscale will lower the trip setpoint, a failure upscale will default to the other flow unit via the auctioneering circuit. The most negative (least positive) auctioneer circuit selects the least positive (smaller of the two amplitudes) of the two flow input signals.

APRM A, C, E are supplied through a low value gate from flow units A and C.

APRM B, D, F are supplied through a low value gate from flow units B and D.

APRMs A, C & E are unaffected by the Flow Unit D failure inasmuch as they are not supplied with flow information from Flow Unit D.

APRMs B, D & F are supplied with auctioneered flow signals from Flow Units B & D. Initially the Flow Unit D signal would be selected by the LVGs. When Flow Unit D fails upscale, the Flow Unit B signal will be selected by the LVGs. Because the Flow Unit B signal is higher than the initial Flow Unit D signal, the trip setpoints for APRMs B, D & F will all rise.

Examination Outline Cross-reference:

(Page 21) 290002 Reactor Vessel Internals K4.02
Knowledge of REACTOR VESSEL INTERNALS design feature(s) and/or interlocks which provide for the following:

- Separation of fluid flow paths within the vessel

Level	RO
Tier #	2
Group #	2
K/A #	290002 K4.02
Importance Rating	3.1

Proposed Question: **# 50**

During power operations, a large through-wall crack in the Reactor Core Shroud has developed.

Which of the following would be a primary indicator in diagnosing the event?

a. A rapid rise in indicated Core Flow with a resulting slight rise in Reactor Power level.

b. The lowering of Jet Pump Differential Pressures on Jet Pumps in the vicinity of the shroud crack.

c. An initial rapid lowering of indicated Reactor Water Level until stabilized by Feedwater Level Control.

d. The cavitation of Recirculation Pumps and/or Jet Pumps due to the heat added by the mixing of fluids.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: GE indicates that a large, through-wall shroud crack will provide indications similar to those experienced at VY (Loose Shroud Head). With a large crack or break in the shroud, enough heat may be added to the downcomer to cause cavitation of the Recirc Pumps, the Jet Pumps, or both.

a. Incorrect: See above. Although indicated core flow would rise due to a lower discharge head pressure on Recirc, reactor power would **lower** due to a higher inlet subcooling temperature.

b. Incorrect: See above. Differential Pressures would **rise** on **all** Jet Pumps with corresponding core flow rise.

c. Incorrect: See above. Shroud failures, depending upon magnitude, may have a corresponding initial **increase** in indicated reactor water level due to the sudden outrush of water; until such time as FWLC reacts to stabilize level.

Technical Reference(s): AOP-32 (Attach if not previously provided)

SDLP-02A/ GE SIL 462

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02A, EO 1.09.a/b/d/e (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
(Page 8) 295034 Secondary Containment Vent Hi Rad EK2.01	Tier #	1	
Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following:	Group #	2	
• Process radiation monitoring system	K/A #	295034 EK2.01	
	Importance Rating	3.9	

Proposed Question: **# 54**

The Reactor Vessel Head has been lifted approximately one (1)-inch in preparation for subsequent removal to support vessel refueling. The Auxiliary Gas Treatment System (AGT) has been discharging to Reactor Building Ventilation for approximately 6 hours when dilution air to the filtration unit is inadvertently reduced. Over the next 15 minutes, Reactor Building Ventilation airborne radioactivity levels climb to 1.5×10^4 cpm in the discharge plenum.

Which of the following actions will occur?

a. EOP-5, Secondary Containment Control will be entered as the result of a Safety Parameter Display System (SPDS) Alarm on EPIC

b. Operators will verify automatic isolation of Reactor Building Ventilation and the automatic initiation of the Standby Gas Treatment System (SGT)

c. Refuel Floor personnel will direct the immediate lowering of the vessel head to avoid exceeding the setpoint for 09-3-2-40 "RX BLDG VENT MON HI-HI"

d. Alarm 09-3-2-40 "RX BLDG VENT MON HI-HI" is received and AGT will automatically trip until discharge effluent can be manually re-routed to SGT

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: At the alarm setpoint for "RX BLDG VENT MONITOR HI-HI" (1.0×10^4 cpm, **automatic** actions occur to isolate Reactor Building Ventilation, start Standby Gas Treatment, and auto-swap AGT from Rx Bldg Ventilation to SGT.

a. Incorrect: JAF is in cold shutdown, Mode 5. EP-1, "EOP Entry and Use" indicates that the "procedure applies during all operating modes, except when reactor coolant temperature is less than 212° F and a reactor startup or shutdown is **not** in progress."

c. Incorrect: This action may, in fact, occur; but, the distracter indicates that action will be taken "prior" to exceeding the setpoint, when the setpoint has been exceeded.

d. Incorrect: Although the alarm will be received, the fact of stating that the "AGT **automatically** trips until discharge effluent can be **manually** re-routed to SGT" makes the statement incorrect as AGT has **no** automatic trip on radiation levels and **automatic** action will occur to swap AGT discharge effluent from Rx Bldg Ventilation to SGT.

Technical Reference(s): OP-31 (Attach if not previously provided)

ARP 09-3-2-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-63, EO 1.05.c.1 & 1.14 (As available)

SDLP-17, EO 1.05.c & 1.14.d.11

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 22) G2.2.1
Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity.

Level	RO
Tier #	3
Group #	[redacted]
K/A #	Cat 2 G2.2.1
Importance Rating	3.7

Proposed Question: **# 55**

A plant startup is in progress with the Mode Selector Switch in "Startup". Control rods are being withdrawn. The Rod Worth Minimizer (RWM) has just failed with 25% of the control rods withdrawn.

What actions are required?

a. Bypass the RWM, verify all further control rod movements are in compliance using a qualified person, and continue the reactor startup.
b. Suspend withdrawal of the control rods, manually SCRAM the reactor, and verify operability of the RWM before commencing a reactor startup.
c. Bypass the RWM, fully insert all control rods, and verify operability of the RWM before commencing a reactor startup.
d. Suspend withdrawal of the control rods, verify operability of the Rod Block Monitor, and continue the reactor startup.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT – Per T.S. 3.3.2.1 and OP-64, Section E.1, if the RWM becomes inoperable below the Low Power Setpoint (LPSP) – reference 25% rods withdrawn in stem. These are the required actions specified in the procedure.
b. Incorrect – Per T.S. Required Action C.1, continued rod movement is suspended except by Scram.
c. Incorrect –Correct only in the "Bypass RWM" portion of the distracter. Per OP-64 and T.S. 3.3.2.1, Control Rod insertion is not required.
d. Incorrect – Although both the RBM and RWM appear in T.S. 3.3.2.1, they serve distinctly different functions and are not backups to each other.

Technical Reference(s): OP-64, Section E.1 / OP-65 (Attach if not previously provided)

T.S. 3.3.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-3D, EO 1.15.a & 1.19 (As available)

Question Source: Bank # 25683

Modified Bank # New

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 15) 261000 SGTS A1.01

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including:

- System flow

Level	RO
Tier #	2
Group #	1
K/A #	261000 A1.01
Importance Rating	2.9

Proposed Question: **# 56**

Following a significant I-131 release in the Reactor Building, Standby Gas Treatment System (SGT) Train "Alpha" has been in service, on an isolated Reactor Building, for several hours and is in the process of being swapped to SGT Train "Bravo". During the swap, the NPO notes Train "B" flow at 5650 scfm. In an attempt to raise flow, the NPO closes valves 01-125SGT-3A/B "SGT Train A/B Fan 1A/B Suction Cross-Tie Isolation Valves". Resulting SGT Train "B" flow is now 5800 scfm.

Which of the following are the potential consequences of the NPO's actions?

a. No consequences; the system will still function as designed
b. Commence drawing a vacuum on the Standby Gas Treatment Room
c. Exceeding the charcoal auto-ignition temperature on SGT Train "Alpha"
d. Excessive SGT Train "Bravo" flows will reduce it's Iodine removal capability

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Closing the 01-125SGT-3A/B Cross-connect valves will secure all decay heat cooling flow from the off-service Train A. With the decay heat buildup, the auto-ignition temperature of the charcoal may be exceeded.

a. Incorrect: There is a potential consequence, as stated above and the "Alpha" train has a high potential for overheating in this case.

b. Incorrect: Closing the 01-125SGT-3A/B Cross-connect valves will secure all decay heat cooling flow from the off-service Train A. Drawing a vacuum on the SGT Room is prevented by the orifice in the decay heat suction line. Without decay heat flow, there is no potential for drawing a vacuum on the SGT Room.

d. Incorrect: The highest flow noted in the stem is 5800 scfm. The maximum flow, by OP-20, is allowed to be as high as 6000 scfm.

Technical Reference(s): OP-20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: FM-48A

Learning Objective: SDLP-01B, EO 1.05.b.6 & 1.13 (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
(Page 13) 217000 RCIC A2.01	Tier #	2	
Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	Group #	1	
	K/A #	217000 A2.01	
• System initiation signal	Importance Rating	3.8	

Proposed Question: **# 58**

The Reactor Core Isolation Cooling System (RCIC) steam supply line is in the process of being tagged out to support corrective maintenance on the Turbine Building Equipment Drain Sump. The following has occurred: (All other components are in their normal positions)

- Air has been isolated to both Steam Line Drain Isolation Valves (13AOV-34/35) and the Drain Trap Bypass Valve (13AOV-32). All are "CLOSED" and tagged in this position.
- Valve Switches for the RCIC Inboard and Outboard Steam Supply Isolation Valves (13MOV-15/16) have just been taken to "CLOSE" by an operator, in preparation for hanging tags. Both valves are "CLOSED"; still have electrical power, and tags have not yet been hung.

Just as 13MOV-15/16 indicate "CLOSED", a feedwater transient results in reactor water level dropping to 126 inches. Without operator action(s), RCIC will...

a. **start automatically with the only subsequent operator action being to start up RHR Torus Cooling as soon as practicable, per Section "Delta" of OP-13B "RHR – Containment Control".**

b. NOT start until all tags are cleared per EN-OP-102 "Protective and Caution Tagging" and all MOVs and AOVs are returned to their "Normal" positions per OP-19 "RCIC" Valve lineup (Attachment 1).

c. attempt to start automatically, but will NOT be able to get steam to the turbine until Inboard / Outboard Steam Supply Isolation Valves (13MOV-15/16) are manually opened per Section "Golf" of OP-19 "RCIC".

d. attempt to start automatically, but will NOT start until an operator takes the 13MOV-16 AUTO CONTROL BYPASS keylock switch (13S-2A) to "BYPASS" per AOP-15 "Isolation Verification and Recovery".

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Per OP-19, Even though 13MOV-15/16 are "normally open", they both still receive an "open" signal on RCIC initiation signal. The only operator action specified on auto-initiation (G.7) is to start RHR Torus Cooling as soon as practicable. Both the Drain Trap Isolation AOV's would close on RCIC initiation directly from the opening of 13MOV-131.

b. Incorrect: See above. RCIC auto-initiation is **not** dependent upon performing any of these actions.

c. Incorrect: See above. Manual opening of 13MOV-15/16 would not be required as they receive an "open" signal upon RCIC initiation signal.

d. Incorrect: See above. By taking the 13MOV-16 AUTO CONTROL BYPASS switch to "BYPASS" it would remove the auto-opening feature and the valve would stay closed; and no steam would reach the turbine.

Technical Reference(s): OP-19 / OP-13B (Attach if not previously provided)

SDLP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO 1.05.a.11/a.17/c.3 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 14) 223002 PCIS/Nuclear Steam Supply Shutoff G2.3.9

Knowledge of the process for performing a containment purge.

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002 G2.3.9	
Importance Rating	2.5	

Proposed Question: **# 61**

An unisolable Main Steam Line Rupture inside Primary Containment has resulted in:

- Emergency Depressurization of the Reactor Pressure Vessel (RPV) with RPV level currently at -5 inches
- Torus Pressure is swiftly approaching the Primary Containment Pressure Limit
- Current radiation levels inside the Drywell are noted on Panel 09-10 as being 190 Rem/Hour.

If all equipment functioned as designed, to initiate a containment Vent and Purge utilizing the "preferred" method, operators will have to...

a. take the Div I and Div II Isolate Switches to "RESET" on the 09-4 / 09-3 respectively, open Torus Exhaust Inner and Outer Isolation Valves (27AOVs), and line up one Containment Atmosphere Dilution (CAD) Train to the Drywell.

b. use at least nine (9) keylock switches on the 27PCP or 27CAD panels to override Group II isolations, open Drywell Exhaust Inner and Outer Isolation Valves (27AOVs), and line up one Containment Atmosphere Dilution (CAD) Train to the Torus.

c. take the Div I and Div II Isolate Switches to "RESET" on the 09-4 / 09-3 respectively, open Drywell Exhaust Inner and Outer Bypass Valves (27MOVs), and line up one Containment Atmosphere Dilution (CAD) Train to the Torus.

d. **use at least nine (9) keylock switches on the 27PCP or 27CAD panels to override Group II isolations, open Torus Exhaust Inner and Outer Bypass Valves (27MOVs), and line up one Containment Atmosphere Dilution (CAD) Train to the Drywell.**

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Stem conditions have at least two Div II signals present (RPV level, DW Press). EP-6 Directs overriding all eight channel inputs on the 27PCP and either A or B Train on the 27CAD to vent from the Torus, using MOVs initially, and Purge through the Drywell (preferred paths).

a. Incorrect: See above. Unless manual isolation directed, which is not a given, these switches are in "RESET" always; AOVs versus MOVs

b. Incorrect: See above. Opposite paths and AOVs versus MOVs

c. Incorrect: See above. Opposite paths.

Technical Reference(s): EOP-4 / EP-6 (Attach if not previously provided)

SDLP-16B/16C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16B, EO 1.05.a.2.c/e & 1.05.b.3 (As available)

SDLP-16C, EO 1.05.a.7/8 & 1.11.b.2/3

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 16) 263000 DC Electrical Distribution K1.04

Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following:

- Ground detection

Level	RO
Tier #	1
Group #	1
K/A #	263000 K1.04
Importance Rating	2.6

Proposed Question: **# 63**

The plant is operating at 100% with operators performing AOP-22, "DC Power System A Ground Isolation," due to a ground on the "Alpha" station battery. The next breaker to be opened is the supply for the 10700 bus breaker Control Power. When this breaker is OPENED, which one of the following statements correctly describes the effect that this will have on the 10700 bus?

a. All 10700 bus breakers will open if originally closed due to a loss of "A" 125 VDC control power.

b. All 10700 bus breaker protection trips will operate normally because bus logic power has automatically swapped to "B" 125 VDC.

c. **All 10700 bus breakers will lose red / green position indicating lights because the breakers have lost "A" 125 VDC control power.**

d. All 10700 bus breaker red / green position indicating lights will still indicate breaker positions because the logic power has automatically swapped to "B" 125 VDC.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT – Per AOP-45/46, a loss of DC power will result in a loss of control power to the affected bus breaker(s) and electrical load indicating lamps.

a. Incorrect – The breakers will not automatically open on loss of DC control power; they fail as-is on a loss of control power.

b. Incorrect – There is no automatic swap of 125 VDC control power on bus 10700.

d. Incorrect – There is no automatic swap of 125 VDC control power on bus 10700, and the red and green lights will be lost when 125 VDC is lost.

Technical Reference(s): AOP-22 / 45 / 46 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B, EO 1.09.a.18 & 1.14a/b/c/d (As available)

Question Source: Bank # 21357

Modified Bank # New

(Note changes or attach parent)

Question History: Last NRC Exam LOI-01-01 (11/5/01)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 16) 264000 EDGs K3.03

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on the following:

- Major loads powered from electrical buses fed by the emergency generator(s)

Level	RO
Tier #	2
Group #	1
K/A #	264000 K3.03
Importance Rating	4.1

Proposed Question: **# 65**

Emergency Diesel Generator (EDG) start signals resulted from a Loss of Coolant Accident concurrent with Degraded 10500/10600 Bus Voltage conditions. The "Alpha" EDG failed to start. All other equipment operates as designed.

Without any operator action, which of the following combinations of Residual Heat Removal (RHR) and Core Spray (CSP) Pumps will be operating 45 seconds after the initial EDG start signals?

a. ONLY "Delta" RHR Pump is running and "Bravo" Core Spray Pump is about to start.

b. **"Alpha, Charlie, and Delta" RHR Pumps along with "Alpha and Bravo" Core Spray Pumps are running.**

c. "Alpha, Bravo, and Delta" RHR Pumps are running with "Alpha and Bravo" Core Spray Pumps about to start.

d. ONLY "Delta" RHR Pump and "Bravo" Core Spray Pump are currently running.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: Upon failure to start of one EDG (in a pair), the 2nd RHR Pump on that side will not auto-start. The rest is just a timing sequence with "A and D" RHR Pumps starting at 11 seconds; "B and C" RHR Pumps at 16 sec ("B" won't start – one EDG); and "A and B" Core Spray Pumps at 21 seconds.

a. Incorrect: See above. Assumes neither "A or C" EDGs started and either "B or D" EDG failed to start also. Also, plays on the 45 second sequence associated with a Degraded Voltage signal without a LOCA signal present.

c. Incorrect: See above. Plays on the power supply oddity of RHR and the 45 second sequence.

d. Incorrect: See above. Assumes neither "A or C" EDGs started and either "B or D" EDG failed to start also.

Technical Reference(s): OP-13A / OP-22 (Attach if not previously provided)

SDLP-10, Att. III

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO 1.03.a, 1.05.a.1.a, 1.06.a, & 1.10.a (As available)

SDLP-93, EO 1.03.a/b, 1.05.b.1/2/3, 1.06.c, & 1.09.b/c

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 20) 272000 Radiation Monitoring A2.16

Ability to (a) predict the impacts of the following on the RADIATION MONITORING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Instrument malfunctions

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	272000 A2.16	
Importance Rating	2.7	

Proposed Question: **# 66**

At 1500, the OPER-INOP keylock switch for 17RM-150A ("Alpha" Off Gas Radiation Monitor) is placed in "INOP" due to a bug source malfunction.

At 1510, 17RM-150B ("Bravo" Off Gas Radiation Monitor) fails downscale.

When will Off Gas Rad Timer (17RM-157) "time out" and which of the subsequent event sequences will occur?

a. At 1515, 17RM-157 times out and AOP-31, "Loss of Condenser Vacuum" is entered to combat the degrading main condenser vacuum

b. At 1515, 17RM-157 times out and AOP-3, "High Activity in Reactor Coolant or Off-Gas" is entered upon closure of 01-107AOV-100 "Off Gas Disch to Stack"

c. At 1525, 17RM-157 times out and AOP-15, "Isolation Verification and Recovery" is entered to recover the Off Gas discharge flowpath

d. At 1525, 17RM-157 times out and AOP-1, "Reactor Scram" is entered upon manual scram due to closure of 01-107AOV-100 "Off Gas Disch to Stack"

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Per ARP 09-3-2-17, *both* rad monitors must fail downscale/inop to start the 15-minute timer. Thus, the timer will not start until time 1510 and will not time out until 1525. At time 1525, 01-107AOV-100 will close on the timer timing out, and ARP 09-3-2-17 directs a manual scram on valve closure.

a. Incorrect: See above, which invalidates time of valve closure on 01-107AOV-100; and, thus, any degrading vacuum condition. AOP-31 could potentially be entered for, or in anticipation of, a degrading condenser vacuum; but, the distracter indicates that degradation is in progress.

b. Incorrect: With no "**upscale** failure", Off Gas Hi Rad, or Off Gas Hi-Hi Rad condition present, AOP-3 would not be entered. Additionally, the above justification invalidates time of valve closure on 01-107AOV-100

c. Incorrect: Time out reference time is accurate from the perspective of 15 minutes after second downscale. AOP-15 deals with several radiation-related plant isolations, but **not** the one in question.

Technical Reference(s): ARP 09-3-2-17 (Attach if not previously provided)

AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01A, EO 1.05.c.1 & 1.10.b (As available)

SDLP-17, EO 1.05.a.4/c.4

Question Source: Bank # [redacted]

Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 8) 295035 Secondary Containment High Differential Pressure/5 G2.1.21

Ability to obtain and verify controlled procedure copy.

Level	RO
Tier #	1
Group #	2
K/A #	295035 G2.1.21
Importance Rating	3.1

Proposed Question: **# 67**

A Reactor Water Cleanup System (RWCU) high energy line break has occurred. Given the following resulting conditions, you have determined the Standby Gas Treatment (SGT) system has failed to automatically initiate:

- The highest RWCU Pump Room general area dose rate is 26 mrem/hr
- "Bravo" RWCU Pump Room Local Area Temperatures are 130°F and 132°F
- Reactor Building Ventilation Exhaust Radiation Levels are 1020 cpm
- Reactor Building Differential Pressure is 1.25 inches Water Gage

The control room hardcopies of procedures conflict with your determination. Which of the below choices would be correct in resolving the conflict?

a. Review the most current revision of AOP-15, "Isolation Verification and Recovery", located in the Technical Reference Center Master Files.

b. **Review the most current revision of OP-20, "Standby Gas Treatment System", located on the Merlin Reference Library Website.**

c. Initiate a Procedure Change Request (PCR) per AP-02.04, "Control of Procedures", to correct the errant procedure(s).

d. Submit a Work Order Request (WOR) per AP-10.01, "Problem Identification and Work Control" to repair the SGT initiation logic.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: The Master Copy of all Operating Procedures is the Merlin Reference Library Website per EN-AD-103 Step 7. This copy of OP-20 will also conflict with the INCORRECT assessment that SGT failed to automatically start.

a. Incorrect: Although AOP-15 will also conflict with the INCORRECT assessment, the Master Copy is NOT located in the Technical Reference Center

c. Incorrect: The procedure is not in error and therefore a Procedure Change Request will not resolve the conflict.

d. Incorrect: The Standby Gas Treatment System is not malfunctioning; and therefore, there is no need to initiate a WOR.

Technical Reference(s): OP-20 (Attach if not previously provided)

AP-12.03

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAP, EO 6.08, 46.04. (As available)

SDLP-01B, EO 1.05.c

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 19) 226001 RHR/LRCI: CTMT Spray Mode K5.06

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI CONTAINMENT SPRAY SYSTEM MODE:

- Vacuum breaker operation

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	226001 K5.06	
Importance Rating	2.6	

Proposed Question: **# 68**

If a small-break Loss of Coolant Accident were to occur in the Drywell, what affect would stuck-open Torus-to-Drywell Vacuum Breaker(s) have on the use of the Drywell and Torus Sprays as compared to the same event with functional Torus-to-Drywell Vacuum Breakers?

With one or more Torus-to-Drywell Vacuum Breakers stuck open, _____ would have to be initiated earlier in the event timeline.

a. ONLY the Torus Spray
b. ONLY the Drywell Spray
c. NEITHER the Drywell nor the Torus Sprays
d. BOTH the Drywell and the Torus Sprays

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – During a small-break LOCA, if the aforementioned vacuum breaker(s) were stuck open, the pressure suppression function of the Torus itself would be degraded or bypassed entirely. Thus, pressures in both the Drywell and Torus air space would rise faster when compared to the same event with fully functional vacuum breakers; requiring initiation of Drywell and Torus Sprays earlier in the progression of events.

a. Incorrect – See above.

b. Incorrect – See above.

c. Incorrect – See above.

Technical Reference(s): SDLP-16A (Attach if not previously provided)

UFSAR, Sections 6.5 and 14

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A, EO 1.05.c.1 & 1.06.b/c (As available)

Question Source: Bank # 25183 (LaSalle 1 – 5/23/2003)

Modified Bank # (Note changes or attach parent)

New

Question History: 1st NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments: *This question was an SRO-Level question on the previous exam and is being used as an RO-Level question on this exam.*

Examination Outline Cross-reference:	Level	RO	SRO
(Page 21) 290003 Control Room HVAC A2.03	Tier #	2	
Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	Group #	2	
	K/A #	290003 A2.03	
• Initiation/reconfiguration failure	Importance Rating	3.4	

Proposed Question: **# 71**

Control Room and Relay Room Air Handling Units (AHUs) are in the process of being swapped to equalize run times on the units. The current configuration is as follows:

- Control Room 70AHU-3A was swapped to 70AHU-3B approximately thirty (30) minutes ago
- An NPO has just started the "Bravo" Relay Room AHU (70AHU-12B) in completing the swap over from "Alpha"

Shortly thereafter, the NPO calls you in the Control Room and reports that 70AHU-12B has tripped. The NPO further reports that 70AHU-12A started from standby and appears to be functioning normally. After acknowledging the NPO, you should direct the NPO to...

a. verify that the running Relay Room Exhaust Fan flow is at least 600 – 900 scfm higher than 70AHU-12A flow in accordance with OP-56 "Relay Room Ventilation and Cooling."

b. **swap the Control Room AHUs back to their previous configuration in accordance with OP-55B "Control Room Ventilation and Cooling" to limit partial loading of the chillers.**

c. reset the High Vibration Trip (commonly received on startup), re-start the "Bravo" Relay Room AHU (70AHU-12B), and secure the "Alpha" AHU in accordance with OP-56 "Relay Room Ventilation and Cooling."

d. start the standby Control Room AHU (70AHU-3A), so that both Control Room AHUs are running; to expeditiously restore the positive pressure to the Control and Relay Rooms in accordance with AOP-73 "Loss of Main Control Room Boundary."

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: A distinct NOTE is provided in both the Control and Relay Room HVAC procedures to run "Alpha" Control and Relay Room AHUs together and "Bravo" Control and Relay Room AHUs together to limit partial loading on the chill water refrigeration units, which could cause spurious trips of the chillers.

a. Incorrect: In the case of the Control and Relay Room Ventilation System lineups, the Exhaust Fans are set to run approximately 600 – 900 scfm **less** than the Supply Fans to maintain a positive pressure in these spaces. Other building ventilation systems do maintain higher exhaust flows.

c. Incorrect: A high vibration condition will not trip the AHUs. It is alarm and white indicating lamp only on ventilation fans.

d. Incorrect: AOP-73 focuses on starting CREVAS and does not support the proposed corrective action detailed in this distracter.

Technical Reference(s): OP-55B (Attach if not previously provided)

OP-56

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70, EO 1.06.a, & 1.13.b/c (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 23) G2.4.26
Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.

Level	RO
Tier #	3
Group #	
K/A #	Cat 4 G2.4.26
Importance Rating	2.9

Proposed Question: **# 72**

Given the following conditions:

- You are responding to an electrical fire as a member of the plant's fire brigade team.
- You have brought a Class B/C fire extinguisher to the scene.
- Other members have rigged a fire hose with a solid-stream nozzle.

Which one of the following actions should be taken?

a. Do **NOT** use the Class B/C fire extinguisher. Put the fire out with the fire hose.
b. Do **NOT** use the fire hose. Put the fire out with the Class B/C fire extinguisher.
c. Use the fire hose first. If it does not put out the fire, use the Class B/C fire extinguisher.
d. Wait for the fire brigade member assigned to bring a Class D fire extinguisher; then use the Class D fire extinguisher.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT – SDLP-76 discusses the use of portable fire extinguishers and what types of fires that they are used on. It also discusses the fact that solid streams are not to be used to fight electrical fires.

a. Incorrect – See above.

c. Incorrect – See above.

d. Incorrect – See above.

Technical Reference(s): SDLP-76 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76, EO 1.05.a.13 (As available)

Question Source: Bank # 25691

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 8) 295029 High Suppression Pool Wtr Lvl EA1.01

Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL:

- HPCI: Plant-Specific

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295029 EA1.01	
Importance Rating	3.4	

Proposed Question: **# 73**

**** Attach copy of Tech Spec 3.6.2.2(NO BASES) ****

The plant is operating at 100% power. This morning, multiple runs of HPCI have been required to establish operability following maintenance on the turbine. The following plant conditions have existed during this time period:

- At 0745, Torus Water Level was monitored at 13.96 feet
- At 0800, Torus Water Level was monitored at 14.01 feet
- At 0830, Torus Water Level was monitored at 14.19 feet
- At 0900, Torus Water Level was monitored at 14.30 feet
- At 0930, Torus Water Level was monitored at 14.42 feet
- At 1000, Torus Water Level was monitored at 14.51 feet

At 1000, a report of which of the following is expected?

a. That HPCI suction valves have automatically swapped to the Torus

b. That the plant has to be in Mode 3, Hot Shutdown, by 2200 (10PM) tonight

c. That RCIC and HPCI suction valves have automatically swapped to the Torus

d. That we now must start physically logging Torus Water Level every 5 minutes

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: HPCI suction valves automatically transfer from the CSTs to the Torus on High Torus Level of 14.5 feet per the ARP 09-3-3-5.

b. Incorrect: Tech Spec LCO 3.6.2.2 has a Note that allows Torus Water Level to be outside the LCO for up to 4 hours

c. Incorrect: HPCI suction valves automatically transfer from the CSTs to the Torus on High Torus Level of 14.5 feet per the ARP 09-3-3-5. RCIC suction valves do NOT have a High Torus Water Level auto-swap design feature.

d. Incorrect: Tech Spec and AOP-36 requirements to monitor / log (respectively) Torus Water TEMPERATURE every 5 minutes, when Torus Water Temperature Recording is inoperable during evolutions which add heat to the Torus.

Technical Reference(s): OP-15 (Attach if not previously provided)

TS 3.6.2.2

Proposed references to be provided to applicants during examination: TS 3.6.2.2 (No Bases Necessary)

Learning Objective: SDLP-23, EO 1.05.a.2/b.1 & 1.06.e (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 5) 295031 Reactor Low Water Level EA2.02

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:

- Reactor power

Level	RO
Tier #	1
Group #	1
K/A #	295031 EA2.02
Importance Rating	4.0

Proposed Question: **# 74**

Following the trip of one (1) Recirculation Pump from 100% steady-state reactor power conditions, which of the following sets of indications would be expected without operator action?

	Reactor Power	INDICATED Reactor Vessel Water Level	Vessel Water Level INSIDE SHROUD
a.	Approximately 52%	Upward surge followed by return to normal band	Downward surge and stabilizes at lower level
b.	Approximately 52%	Upward surge with sustained high level	Upward surge with sustained high level
c.	Approximately 65%	Upward surge followed by return to normal band	Downward surge and stabilizes at lower level
d.	Approximately 65%	Upward surge with sustained high level	Upward surge with sustained high level

Proposed Answer: **c**

Explanation (Optional):

c. CORRECT: Per AOP-8, trip of one Recirc Pump yields a power value of approximately 65%. Symptoms further dictate an upward surge in indicated level. Further tests theory of level effects inside shroud and the impact on power level. Increased core voiding displaces water inside shroud and drives power down.

a. Incorrect: See above. 52% is the value for trip of **both** Recirc Pumps. Remainder is correct for conditions.

b. Incorrect: See above. 52% wrong. Also, FWLC would act to restore indicated level. Water level inside shroud would not go up. Tests misconception on inside versus indicated level.

d. Incorrect: Correct power, but FWLC would act to restore indicated level and water level inside shroud would not go up.

Technical Reference(s): AOP-8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H, EO 1.10.e, 1.12.a, & 1.15.a (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 6) 600000 Plant Fire On Site G2.1.22
Ability to determine Mode of Operation.

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	600000 G2.1.22	
Importance Rating	2.8	

Proposed Question: **# 75**

A reactor startup is in progress, following a refuel outage, with the following plant conditions:

- IRMs on mid-scale on Range 7
- Average Reactor Coolant Temperature is currently 200° F and rising slowly

A Zone RB-1W (Reactor Building West Crescent) Fire Alarm is received. An NPO is quickly dispatched to the location and reports significant heat, smoke, and flames. All AOP-28, "Operation During Plant Fires" prompt operator actions are completed satisfactorily. Which of the following accurately reflects the plant's current "Mode of Operation"?

a. Mode 2: Startup / Hot Standby
b. Mode 3: Hot Shutdown
c. Mode 4: Cold Shutdown
d. Mode 5: Refueling

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Initial conditions place the plant in Mode 2. With a confirmed fire in Zone RB-1W, a Reactor Scram would be inserted and AOP-1 will direct taking the Mode Switch to "Shutdown"; resulting in either Mode 3 or 4 operation. With resulting Average Reactor Coolant Temperature below 212° F, Mode 4 operation would result.

a. Incorrect: Plant in Mode 2 currently. If candidate fails to recognize AOP-28 Prompt Action Reactor Scram, this would be the resulting mode.

b. Incorrect: See above. With resulting Average Reactor Coolant Temperature below 212° F, Mode 3 operation would **NOT** result.

d. Incorrect: See above. Mode switch can be in "Shutdown" for Mode 5 if vessel head not fully tensioned. If candidate fails to recognize dependency on tensioning, this could be the resulting mode.

Technical Reference(s): AOP-28 (Attach if not previously provided)

ITS (Mode Definitions)

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS02, EO 1.03 (As available)

LPAOP, EO 1.03

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 4) 295026 Suppression Pool High Water Temp. / 5
G2.1.18
Ability to make accurate / clear and concise logs / records / status boards / and reports.

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295026 G2.1.18	
Importance Rating	2.9	

Proposed Question: **# 76**

During the performance of ST-22B, MANUAL SAFETY RELIEF VALVE OPERATION AND VALVE MONITORING SYSTEM FUNCTIONAL TEST, Torus water temperature has steadily risen to a value of 92°F when it is determined that the 16-1TR-131A (09-4 Torus Water Temperature Recorder) has become inoperable.

Which of the below is the required operator action?

a. Secure the test until Torus water temperature is reduced to <85°F.
b. Log Torus water temperature at 15 minute intervals.
c. Secure the test if Torus temperature exceeds 95°F.
d. Observe Torus water temperature at 5 minute intervals.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Per T/S 3.6.2.1, Torus water temperature is limited to 105° during testing that adds heat to the Torus. SR 3.6.2.1.1 requires a 5 minute interval verification of Torus temperature within limitation during this testing

a. Incorrect: See Above.

b. Incorrect: Logging Torus water temperature at 5 minute intervals is an AOP-36 requirement if both 09-4 temperature recorders are inoperable. The conditions do not warrant entry to AOP-36 and 16-1TR-131B is still available.

c. Incorrect: 95° Torus water temperature is the EOP-4 entry condition and the T/S 3.6.2.1 limitation during normal operations which are not adding heat to the Torus. ST-22B and T/S 3.6.2.1 allow operation to 105° in the stem conditions provided.

Technical Reference(s): T/S 3.6.2.1 (Attach if not previously provided)

ST-22B

Proposed references to be provided to applicants during examination: None

Learning Objective: _ (As available)

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 15) 262001 AC Electrical Distribution G2.4.27
Knowledge of fire in the plant procedure.

Level	RO
Tier #	2
Group #	1
K/A #	262001 G2.4.27
Importance Rating	3.0

Proposed Question: **# 77**

Which of the following operator actions would be consistent with preventing a Multiple High Impedance Fault (MHIF) from occurring during a plant fire in a given area?

a. Place individual component control switches for loads in the affected area in "Pull-To-Lock"
b. Transfer safe shutdown loads to alternate components not located in the affected area
c. Trip the individual load breakers for components / panels in the affected area
d. Trip the L-Gear feeder breaker associated with loads in the affected area

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: MHIF, per AOP-28 definition, is a low probability phenomenon whereby several fire induced faulted circuits cause sufficient amperage to trip the panel or MCC feeder breaker before any of the individual load breakers trip. This could result in a loss of power to the entire panel or MCC unless operator action is taken to shed the affected loads.

a. Incorrect: See above. Although this action would secure loads, it does not meet the prescribed manner of AOP-28.

b. Incorrect: See above. Although this action would secure loads, it does not meet the prescribed manner of AOP-28.

d. Incorrect: See above. Overly conservative approach that may de-energize necessary loads needlessly.

Technical Reference(s): AOP-28 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO 1.08 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 5) 295030 Low Suppression Pool Wtr Lvl / 5 EA1.06

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:

- Condensate storage and transfer (make-up to the suppression pool): Plant-Specific

Level	RO
Tier #	1
Group #	1
K/A #	295030 EA 1.06
Importance Rating	3.4

Proposed Question: **# 79**

While operating the RCIC system in the Full Flow Test Mode per ST-24J, RCIC FLOW RATE AND IN-SERVICE TEST, an I/C test results in simultaneous full opening of both RCIC Torus Suction Valves, 13MOV-39 and 41.

Which of the below describes the expected system response?

a. **The RCIC CST Suction valve (13MOV-18) and Full Flow Test valve (13MOV-30) will both close. RCIC will continue to operate on minimum flow only.**

b. The HPCI Full Flow Test CST Return valve (23MOV-24) opens. RCIC continues to operate with return to the CST via HPCI.

c. The RCIC CST Suction valve (13MOV-18) closes. RCIC continues to operate in Full Flow Test.

d. No additional changes. RCIC continues to operate in Full Flow Test.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Either torus suction valve opening will close the full flow test valve to prevent pumping the torus water volume to the CST. When both torus suctions are full open, the CST suction will close thus maintaining RCIC suction during the transfer. The net result is RCIC continues to operate with the full flow test path isolated resulting in min flow valve opening.

b. Incorrect: Although this flowpath is used for RCIC testing, there are no automatic openings to the HPCI full flow test return to the CST and this valve is in series with the RCIC full flow test valve which closes.

c. Incorrect: See Above.

d. Incorrect: See Above.

Technical Reference(s): ST-24J (Attach if not previously provided)

ARP 09-4-0-28

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO-1.06.b, d, 1.12.d, and 1.14.b (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 22) G2.2.24
Ability to analyze the affect of maintenance activities on LCO status.

Level	RO
Tier #	3
Group #	
K/A #	Cat 2 G2.2.24
Importance Rating	2.8

Proposed Question: **# 80**

The Plant is operating at 100% power steady state with HPCI tagged out for maintenance. You are currently in day three (3) of the Technical Specification Required Actions / Completion Times associated with the HPCI LCO. The SNO is performing a surveillance test to demonstrate RCIC operability and reports that the Barometric Compressor Vacuum Pump, (13P-3), will not start.

Which of the below describes appropriate action and the impact on Technical Specifications?

a. Dispatch NPO to check breaker status on BMCC-1/3. No change in LCO.
b. Dispatch NPO to check breaker status on BMCC-2/4. No change in LCO.
c. Dispatch NPO to check breaker status on BMCC-2/4. LCO is more restrictive.
d. Dispatch NPO to check breaker status on BMCC-1/3. LCO is more restrictive.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Tests the recognition of "Alpha" DC to RCIC, whereas "Bravo" DC powers HPCI. Additionally, with HPCI inop, RCIC must be operable. With the operability of RCIC called into question on the failure to start of the vacuum pump, action must be taken immediately to determine / verify RCIC status.

a. Incorrect – See above.

b. Incorrect – See above.

c. Incorrect – See above.

Technical Reference(s): OP-19 (Attach if not previously provided)

OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO 1.04.a & 1.16 (As available)

SDLP-23, EO 1.16

Question Source: Bank # 25649

[redacted] (Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments: *This question was an SRO-Level question on the previous exam and is being used as an RO-Level question on this exam.*

Examination Outline Cross-reference:

(Page 5) 295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown EK3.01

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:

- Recirculation pump trip/runback: Plant-Specific

Level	RO
Tier #	1
Group #	1
K/A #	295037 EK3.01
Importance Rating	4.1

Proposed Question: **# 82**

Which of the following is the basis for the ATWS Trip of the Recirculation Pumps?

Tripping the Recirculation Pumps...

a. causes indicated reactor level to go down which results in a rapid power reduction.

b. adds a large amount of negative reactivity quickly by increasing core inlet subcooling.

c. causes faster control rod scram times, upon lowering core d/p, resulting in quicker power reduction.

d. **adds a large amount of negative reactivity quickly due to the resulting large rise in core void fraction.**

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Per TS Bases 3.3.4.1, on ATWS RPT Instrumentation, tripping of the Recirculation Pumps adds negative reactivity from the increase in steam voiding in the core area as the core flow decreases.

a. Incorrect – not the basis.

b. Incorrect – not the basis.

c. Incorrect – not the basis.

Technical Reference(s): TS Bases 3.3.4.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H, EO 1.07.a & 1.17.c (As available)

Question Source: Bank # 23563 (Columbia Generating Station 2 3/27/03)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 7) 295007 High Reactor Pressure AK1.01

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:

- Pump shutoff head

Level	RO
Tier #	1
Group #	2
K/A #	295007 AK1.01
Importance Rating	2.9

Proposed Question: **# 84**

The Core Spray System has automatically initiated on Low Reactor Vessel Water Level.

The Core Spray Injection Valve opens when reactor pressure drops below _____.

When the injection valve opens, RPV injection _____ immediately.

a. 310 psig ; occurs
b. 310 psig ; does **NOT** occur
c. 450 psig ; occurs
d. 450 psig ; does **NOT** occur

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT – Per TS 3.3.5.1.c, Core Spray Injection Permissive Allowable value is between 410 psig and 490 psig. Actual RPV injection occurs at approximately 200 psig (RPV). Until such time as this pressure is reached (lowering), the pump will run at shutoff head in relation to injecting to the RPV.

a. Incorrect – See above, incorrect pressure. Teamed with additional incorrect response.

b. Incorrect – See above, incorrect pressure. Teamed with correct response if that were correct permissive.

c. Incorrect – See above, correct pressure. Reactor pressure still too high to actually inject and will not actually inject until well below 300 psig.

Technical Reference(s): OP-14 (Attach if not previously provided)

TS 3.3.5.1.c (and Bases)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14, EO 1.05.a.6/b.1/c.4 & 1.07.b (As available)

Question Source: Bank # 25620

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 21) 290001 Secondary CTMT A2.01
Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- Personnel airlock failure

Level	RO
Tier #	2
Group #	2
K/A #	290001 A2.01
Importance Rating	3.3

Proposed Question: **# 85**

The plant is operating normally at full power. A Spent Fuel Cask is being moved out of the reactor building. While the reactor building track bay outer doors were open, a power supply failure de-energized the track bay door control system.

Which one of the below describes the required operator action(s).

a. Declare Secondary Containment inoperable. Per TS 3.6.4.1, restore secondary containment within 4 hours.

b. Restrict track bay access via the inner doors. Per OP-73, Miscellaneous Plant Equipment, dispatch a NPO to manually close the outer doors.

c. Consider Secondary Containment degraded. Per OP-51A, Reactor Building Ventilation and Cooling System, isolate reactor building ventilation.

d. Declare the Secondary Containment Penetration inoperable. Per TS 3.6.4.2, deactivate the inner doors in the closed condition within 8 hours.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: As long as the inner set or the outer set is closed and sealed, Secondary Containment continues to be operable. The pneumatic seal solenoid is fail safe to maintain the seal pressurized. OP-73 contains actions to operate the doors in the stem described condition.

a. Incorrect: See Above.

c. Incorrect: Although secondary containment is degraded, it is still intact and operable. Isolating reactor building ventilation is not a required action for this condition.

d. Incorrect: See Above. This penetration is not inoperable. The referenced T/S is for Secondary Containment Isolation Valves, not penetrations.

Technical Reference(s): OP-73 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TS 3.6.4.1, 3.6.4.2

Learning Objective: SDLP-16A EO-1.05.a.9 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 15) 262002 UPS (AC/DC) A4.01

Ability to manually operate and/or monitor in the control room:

- Transfer from alternative source to preferred source

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262002 A4.01	
Importance Rating	2.8	

Proposed Question: **# 87**

SEE ATTACHED...

Given the above **"BEFORE"** and **"AFTER"** Alarm Panel indications, determine which of the following events could result in the above change in alarm conditions:

a. A loss of UPS AC Drive has occurred resulting in a UPS auto-swap to DC Drive

b. UPS is in the process of being returned to AC Drive and is currently on DC Drive for synchronization

c. A loss of AC Drive to UPS has occurred and operator action has been taken to swap to backup DC power

d. A fault has occurred in the Alternate AC feed to UPS resulting in a UPS auto-swap to its DC power supply

Proposed Answer: **b.**

Explanation (Optional):

b. Correct: In the "Before" picture above, UPS is already on the Alternate AC Feed due to a low voltage condition on the DC Drive. In the "After" sequence, and minus additional alarms, the "UPS ON DC DRIVE" indicates that the MG Set is in the process of being restored and is on the DC Drive solely for the purpose of synchronization prior to swapping back to the normal AC Drive.

a. Incorrect: Taking the "Before" alarm sequence into account, the MG set AC and DC Drives had been lost and the Alternate AC Feed was supplying loads.

c. Incorrect: Taking the "Before" alarm sequence into account, the MG set AC and DC Drives had been lost and the Alternate AC Feed was supplying loads. Additionally, if AC Drive had been lost, no operator action would be necessary, as there is an auto-swap to DC Drive.

d. Incorrect: There is no auto-swap feature from Alternate AC Feed to DC Drive.

Technical Reference(s): OP-46B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71F, EO 1.05.c & 1.13 (As available)

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

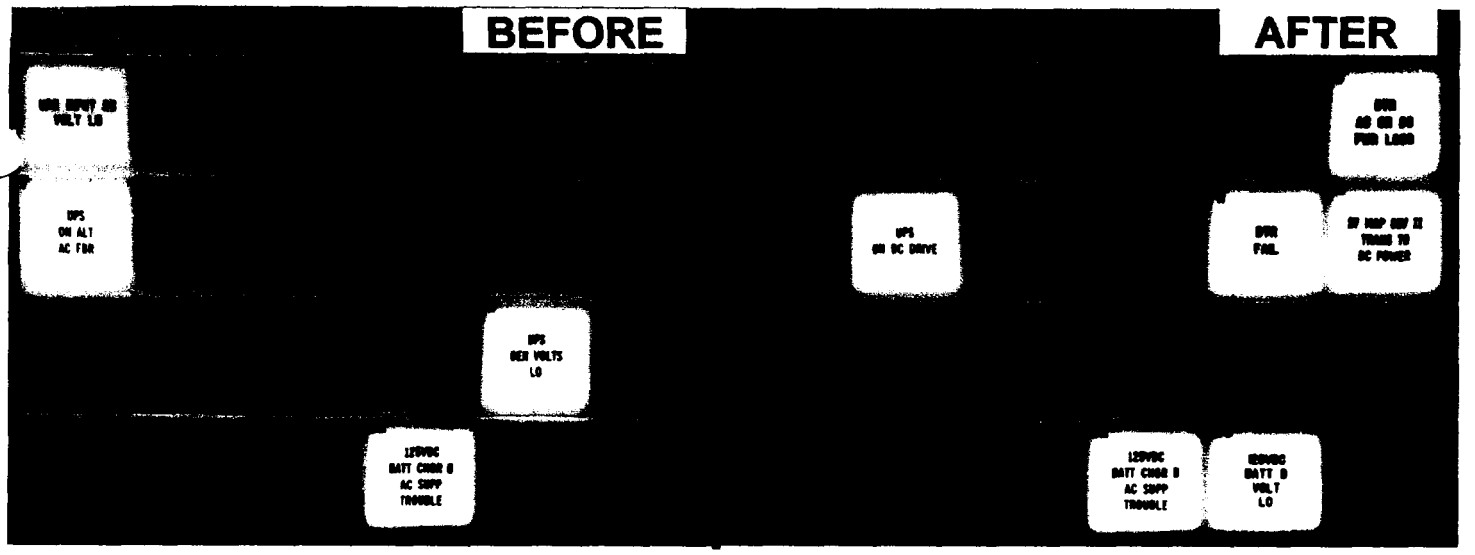
Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:



Given the above **"BEFORE"** and **"AFTER"** Alarm Panel indications, determine which of the following events could result in the above change in alarm conditions:

a. A loss of UPS AC Drive has occurred resulting in a UPS auto-swap to DC Drive

b. UPS is in the process of being returned to AC Drive and is currently on DC Drive for synchronization

c. A loss of AC Drive to UPS has occurred and operator action has been taken to swap to backup DC power

d. A fault has occurred in the Alternate AC feed to UPS resulting in a UPS auto-swap to its DC power supply

b.	Correct: In the "Before" picture above, UPS is already on the Alternate AC Feed due to a low voltage condition on the DC Drive. In the "After" sequence, and minus additional alarms, the "UPS ON DC DRIVE" indicates that the MG Set is in the process of being restored and is on the DC Drive solely for the purpose of synchronization prior to swapping back to the normal AC Drive.
a.	Incorrect: Taking the "Before" alarm sequence into account, the MG set AC and DC Drives had been lost and the Alternate AC Feed was supplying loads.
c.	Incorrect: Taking the "Before" alarm sequence into account, the MG set AC and DC Drives had been lost and the Alternate AC Feed was supplying loads. Additionally, if AC Drive had been lost, no operator action would be necessary, as there is an auto-swap to DC Drive.
d.	Incorrect: There is no auto-swap feature from Alternate AC Feed to DC Drive.

Examination Outline Cross-reference:

(Page 14) 259002 Reactor Water Level Control A4.02
****Recent Plant Modification****

Ability to manually operate and/or monitor in the control room:

- All individual component controllers in the automatic mode

Level	RO
Tier #	2
Group #	1
K/A #	259002 A4.02
Importance Rating	3.7

Proposed Question: **# 88**

The reactor is operating steady-state 100% power when Annunciator 09-5-1-28 "RX WATER LVL ALARM HI OR LO" alarms. As you observe, reactor water level continues to trend downward until level steadies out at +189 inches. Which of the following impacts on Feedwater Level Control is the probable cause?

a. Input from one steam flow signal transmitter has been lost
b. Inadvertent sustained opening of one Safety Relief Valve (SRV)
c. Input from a single feedwater flow signal transmitter has been lost
d. Inadvertent High Pressure Coolant Injection System (HPCI) initiation

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Loss of one steam flow transmitter causes indicated steam flow to drop below feed flow. Actual feed flow decrease to compensate. Since actual steam flow remains at the original rate, an actual mismatch occurs and level drops. A developed level error signal slows the transient, and steady-state is restored when level error exactly compensates for the reduction in indicated steam flow. This requires 12 – 15 inches and final actual level ends up at approximately 190 inches.

b. Incorrect: One SRV yields approximately 8% steam flow and therefore, an 8% mismatch. System anticipates a level transient and biases sensed level to appear 5 inches high (8% of 60). System stabilizes with level 5 inches lower than level set. You would get alarm, but not additional level drop.

c. Incorrect: System sees a 50% mismatch, anticipates transient, and biases level to appear 24-30 inches low. RFP's speed up and level increases to the RFPT and Main Turbine/Reactor Scram setpoints before stabilized.

d. Incorrect: Approximate 3% Steam Flow and 20% Feed Flow. Result is approximate 16% mismatch. System anticipates and biases level to appear 10 inches low. System stabilizes with level 10 inches higher than level set.

Technical Reference(s): AOP-41/42 (Attach if not previously provided)

SDLP-06

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-06, EO 1.08.b & 1.10.e (As available)

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 16) 263000 DC Electrical Distribution A3.01

Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including:

- Meters, dials, recorders, alarms, and indicating lights

Level	RO
Tier #	2
Group #	1
K/A #	263000 A3.01
Importance Rating	3.2

Proposed Question: **# 89**

With normal full-power loading on the system, 125 VDC Battery Bus Voltage (as read on the 09-8 Panel) should be approximately _____; and, in the event that AC input is lost to the "Alpha" 125 VDC Battery Charger, the same 125 VDC BATT BUS A meter would

_____.

a. 132 VDC; steadily lower to approximately 118 VDC, alarming annunciator 09-8-1-20 "125 VDC BATT A VOLT LO" several minutes later.

b. 132 VDC; promptly lower slightly to approximately 128 VDC, alarming annunciator 09-8-1-19 "125 VDC BATT CHGR A AC SUPP TROUBLE immediately.

c. 125 VDC; remain constant at approximately 125 VDC, alarming annunciator 09-8-1-19 "125 VDC BATT CHGR A AC SUPP TROUBLE several minutes later.

d. 125 VDC; promptly lower slightly to approximately 122 VDC, alarming annunciator 09-8-1-20 "125 VDC BATT A VOLT LO" immediately.

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Per OP-43A, Section E, normal float voltage verified by operators shiftly to be 131 – 133 VDC. SDLP-71B specifically mentions the 118 VDC value on loss of charger. ARP 09-8-1-19/20 discuss the two alarms noted above with a specific note in 19 (AC SUPP TROUBLE), which does come in immediately; to expect 20 (BATT A VOLT LO) in approximately 3 minutes on loss of AC input.

b. Incorrect: See above. Correct initial VDC; Wrong Drop; Correct alarm but wrong input.

c. Incorrect: See above. Wrong initial VDC: Wrong response: Wrong alarm in response to timeframe.

d. Incorrect: See above. Wrong initial VDC: Wrong Drop; Wrong alarm in response to timeframe.

Technical Reference(s): OP-43A (Attach if not previously provided)

ARP 09-8-1-19/20

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B, EO 1.05.a.1, 1.06.a, & 1.10.a.1 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 16) 300000 Instrument Air K2.01

Knowledge of electrical power supplies to the following:

- Instrument air compressor

Level	RO
Tier #	2
Group #	1
K/A #	300000 K2.01
Importance Rating	2.8

Proposed Question: **# 90**

The 10300 Bus has been lost. L-13 and L-23 have been re-energized. How many Air Compressors are available?

a. None

b. One

c. Two

d. Three

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Per OP-39, Air compressor power supplies as follows: 39AC-2A / L-23; 39AC-2B / L-24; 39AC-2C / L-33. L-33 has been lost and is yet to be cross-connected; therefore 39AC-2C is unavailable, leaving two (2) available compressors.

a. Incorrect: see above.

b. Incorrect: see above.

d. Incorrect: see above.

Technical Reference(s): OP-39 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39, EO 1.03.a (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 7) 295012 High Drywell Temperature AK2.02
Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following:

- Drywell cooling

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295012 AK2.02	
Importance Rating	3.6	

Proposed Question: **# 91**

Two hours into your shift, you notice that Torus Water Level has risen from 13.91 feet to 13.94 feet while Drywell to Torus Differential Pressure has risen from 1.75 psid to 1.80 psid and Torus pressure has remained constant at 0.0 psig. You have confirmed the indications on EPIC LOG 1.

Which of the following is an appropriate action for these conditions?

a. Per OP-37 "Containment Atmosphere Dilution System", vent the Drywell to return differential pressure to normal band.

b. Per OP-40 "Reactor Building Closed Loop Cooling", raise Service Water flow(s) to in-service RBCLC Heat Exchangers.

c. Enter AOP-9 "Loss of Primary Containment Integrity" and dispatch Operators to search for Primary Containment leakage.

d. Enter EOP-4 "Primary Containment Control" and immediately dispatch Operators to drain water from the Torus to restore level to normal band.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: Stem conditions are symptomatic of a rise in RBCLC temperature. A leak in Primary Containment would cause the reverse of provided stem indications based on a leak from the Torus.

a. Incorrect: See above. Per OP-37, the stem indications of parameters on Drywell to Torus D/P are within the normal band. Additionally, any venting would be from the Torus versus the Drywell.

c. Incorrect: See above. Related symptoms do not exist for entry into AOP-9. For example, Drywell to Torus D/P **lowering** is a symptom for AOP-9 entry and stem conditions have it rising.

d. Incorrect: See above. EOP-4 entry conditions of 13.88 feet or 2.7 psig Drywell Pressure do not exist. Therefore, EOP-4 entry is not warranted.

Technical Reference(s): OP-40 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15, EO 1.05.a.2, 1.09.b, 1.10.a, & 1.13.c (As available)

Question Source: [redacted] Bank # [redacted]

Modified Bank # 25628 (attached) (Note changes or attach parent)

Question History: [redacted] New [redacted] Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
FitzPatrick 1	7/1/2003	ILO	25628	BWR

Two hours into the shift, the SNO reports that Torus water level has dropped from 14.0 ft to 13.91 ft while Drywell to Torus ?P has dropped from 1.8 psid to 1.6 psid and Torus pressure has remained constant at 0.0 psig. You have confirmed the indications on EPIC-LOG1.
Your required actions are.

Answer **Enter AOP-9, Loss of Primary Containment Integrity, and dispatch Operators to search for Primary Containment leakage.**

Distractor 1 Enter EOP-4, Primary Containment Control, and immediately makeup nitrogen to the Drywell restore ?P.

Distractor 2 Enter EOP-4, Primary Containment Control, and immediately makeup water to the Torus to restore Torus level.

Distractor 3 Enter AOP-9, Loss of Primary Containment Integrity, and dispatch Operators to determine why RBCLC temperature has risen.

RefMaterial

QuestionComment "EOP-4 Entry conditions of 13.88 ft or 2.7 psig DW pressure do not exist, therefore EOP-4 Entry is not required.
"Adequate AOP-9 symptoms do exist warranting entry.
"The stem conditions are symptomatic of Primary Containment leakage. A RBCLC temperature rise will cause the reverse of the indications based on higher DW temperatures caused by less heat removal by the DW cooling units.

Distract1 Comment

Distract2 Comment

Distract3 Comment

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
..295030.EA2.04			295030	EA2	04
KaRevision				**CognitiveLevel**	**ExamLevel**
				2	S

Examination Outline Cross-reference:

(Page 8) 295015 Incomplete SCRAM/1 G2.4.20

Knowledge of operational implications of EOP warnings / cautions / and notes.

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295015 G2.4.20	
Importance Rating	3.3	

Proposed Question: **# 92**

A Feedwater transient has resulted in a Reactor Scram on Low Reactor Water Level. The following plant conditions result:

- RPV Water Level is currently stable at 140 inches
- Rod 26-27 remains at position 06 in the core
- Rod 34-35 stops at position 04 in the core
- Rod 18-35 position indication is lost during the scram
- All Intermediate Range Monitors (IRMs) are on Range 6 and trending down slowly

Which of the following must be accomplished?

a. Use AOP-1 to insert Control Rods and *rapidly* restore Reactor Water Level to 177 – 222.5 inches.

b. Use AOP-1 to insert Control Rods and *slowly* restore Reactor Water Level to 177 - 222.5 inches.

c. Use EP-3 to insert Control Rods and *rapidly* restore Reactor Water Level to 177 – 222.5 inches.

d. Use EP-3 to insert Control Rods and *slowly* restore Reactor Water Level to 177 – 222.5 inches.

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Scram gets you into AOP-1. Reactor Water Level less than scram setpoint is EOP-2 entry. With rods out and shutdown not assured without Boron, EOP-3 entry, and thus EP-3, is warranted. Further tests EOP-3 Caution on rapid level changes to prevent subsequent power excursions.

a. Incorrect: See above. **EOP-2 entry** condition would take precedence and drive you to EOP-3 actions. Also, "**rapidly**" would be wrong.

b. Incorrect: See above. Other than "**AOP-1**" portion of distracter, the remainder is correct.

c. Incorrect: See above. Correct with exception of "**rapidly**".

Technical Reference(s): AOP-1 (Attach if not previously provided)

EOP-2 / EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C, EO 1.02 & 1.03 (As available)

MIT-301.11D, EO 1.02 & 1.05

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

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 5) 295028 High Drywell Temperature /5 EK3.04

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE:

- Increased drywell cooling

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295028 EK3.04	
Importance Rating	3.6	

Proposed Question: **# 93**

A Design Basis Loss of Coolant Accident has occurred. In focusing on the Drywell Temperature leg of EOP-4 "Primary Containment Control", which of the following must be accomplished, and why?

a. In accordance with "Operate all available drywell cooling", start the fourth fan for each Drywell Unit Cooler to maximize the convective cooling provided by the operating unit coolers.

b. In accordance with "Operate all available drywell cooling", start the fourth fan for each Drywell Unit Cooler to maximize the evaporative and convective cooling provided by the operating unit coolers.

c. In accordance with "BEFORE Drywell temperature reaches 309° F", initiate Drywell Sprays because convective cooling far surpasses evaporative cooling in providing rapid pressure / temperature reductions.

d. **In accordance with "BEFORE Drywell temperature reaches 309° F", initiate Drywell Sprays because evaporative cooling far surpasses convective cooling in providing rapid pressure / temperature reductions.**

Proposed Answer: **d.**

Explanation (Optional):

d. CORRECT: Per Emergency Procedure Guidelines (EPGs), Evaporative cooling results when the sprayed water droplets flash to steam; absorbing the latent heat of vaporization from the surrounding atmosphere. This process can result in a relatively large drop in Drywell pressure and may occur at a rate faster than can be compensated for by operator action. Alternatively, Convective cooling results when the sprayed water droplets absorb sensible heat from the surrounding atmosphere resulting in a pressure/temperature reduction until equilibrium conditions are established. This process occurs at a rate much slower than evaporative cooling and can be controlled by terminating sprays.

a. Incorrect: See above. The operation of four Drywell Unit Cooler Fans is strictly prohibited, except for short periods to alternate fans, by OP-53 Precautions. Additionally, during a DBA LOCA, all Drywell Fans would trip on high vibration.

b. Incorrect: See above. Also, Unit Coolers do not provide evaporative cooling.

c. Incorrect: See above. Reverses effectiveness of "Convective" versus "Evaporative" cooling effect.

Technical Reference(s): EPGs (Attach if not previously provided)

EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E, EO 4.05 (As available)

Question Source: Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 5) 295038 High Off-Site Release Rate EK2.03
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following:

- Plant ventilation systems

Level	RO
Tier #	1
Group #	1
K/A #	295038 EK2.03
Importance Rating	3.6

Proposed Question: **# 94**

A Loss of Coolant Accident (LOCA) has occurred. The following conditions are present:

Reactor Building Ventilation.....ISOLATED

Reactor Building Ventilation Exhaust Radiation.....1 x 10^5 cpm

"A" Standby Gas Train.....OPERATING

Reactor Building to Atmosphere Differential Pressure..... .negative (-)1.1 inches water

Turbine Building Ventilation .....ISOLATED

Turbine Building Exhaust Radiation.....3 x 10^4 cpm

Offsite Release .....Above the ALERT Level

Which ventilation system should be re-established and why? Re-establish the...

a. Turbine Building Ventilation to filter the ventilation exhaust from the turbine building.
b. Reactor Building Ventilation to prevent an unmonitored ground level release of radioactivity.
c. **Turbine Building Ventilation to prevent an unmonitored ground level release of radioactivity.**
d. Reactor Building Ventilation to reduce the reactor building area and equipment temperatures.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT – The turbine building is not a leak tight building. Restarting the turbine building vent will result in monitored ground releases vice unmonitored ground releases for any radioactivity in the building.

a. Incorrect – There is no filtration on turbine building exhaust.

b. Incorrect – EOP-4 states that if reactor building exhaust radiation is greater than 1x10^4 cpm, then isolate reactor building vents.

d. Incorrect – EOP-4 states that if reactor building exhaust radiation is greater than 1x10^4 cpm, then isolate reactor building vents.

Technical Reference(s): EOP-6 EPG (Attach if not previously provided)

EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11G, EO 6.04 (As available)

Question Source: Bank # 21340

Modified Bank # [redacted]

New [redacted]

(Note changes or attach parent)

Question History: Last NRC Exam LOI-01-01 (11/5/01)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
(Page 14) 239002 SRVs K5.05	Tier #	2	
Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES:	Group #	1	
• Discharge line quencher operation	K/A #	239002 K5.05	
	Importance Rating	2.6	

Proposed Question: **# 95**

While experiencing Torus water level control problems, an operator opens a Safety Relief Valve (SRV) with Torus water level at 5.2 ft.

Opening the SRV under these conditions will result in...

a. drawing water up into the SRV Tailpipe.
b. direct Suppression Chamber pressurization.
c. valve seat damage from excessive SRV flow rates.
d. excessive hydrodynamic loading on the SRV Tailpipe.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT – EOP EPG Bases indicates that SRV Tailpipe T-Quenchers become uncovered at 5.5 feet and discuss the resulting direct Torus pressurization. Tech Spec Bases 3.6.2.2 mirror this discussion.

a. Incorrect – associated with failure of vacuum breaker(s).

c. Incorrect – not an analyzed design issue.

d. Incorrect – associated with Torus water level too **high**, versus low.

Technical Reference(s): EOP-2/3, and EPG Bases (Attach if not previously provided)

TS Bases 3.6.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A, EO 1.05.a.7.e, 1.09.f, & 1.10.c (As available)

Question Source: Bank # 25614

Modified Bank # New (Note changes or attach parent)

Question History: Last NRC Exam LOI-03-01 (7/1/03)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 17) 400000 Component Cooling Water K4.01

Knowledge of CCWS design feature(s) and/or interlocks which provide for the following:

- Automatic start of the standby pump

Level	RO
Tier #	2
Group #	1
K/A #	400000 K4.01
Importance Rating	3.4

Proposed Question: **# 96**

The Service Water System is in a normal lineup with 46P-1A and 46P-1C running and 46P-1B in standby. Alarm 09-6-2-3 "SERV WTR PMP 46P-1A OVERLOAD OR TRIP" annunciates.

Which of the following is expected?

a. Service Water Pump 46P-1B has auto-started on the trip of 46P-1A
b. Service Water Pump 46P-1B must be manually started at 09-6 Panel
c. Manually scram the reactor and manually trip both Recirculation Pumps
d. Service Water Pump 46P-1B has auto-started on low NSW header pressure

Proposed Answer: **a.**

Explanation (Optional):

a. CORRECT: Per ARP 09-6-2-3, if 46P-1A trips, the standby pump will start. With 46P-1B in standby, it will auto-start on trip of 46P-1A.

b. Incorrect: Per ARP 09-6-2-3, if 46P-1A trips, the standby pump will start **automatically**. With 46P-1B in standby, it will **auto-start** on trip of 46P-1A.

c. Incorrect: The scram and tripping of the Recirc pumps is a Prompt Action of AOP-10, "Loss of Service Water" during a **complete** loss of Service Water. The stem conditions do not support a **complete** loss.

d. Incorrect: Per ARP 09-6-2-3, if Service Water header pressure lowers to less than **75** psig, the standby pump will start. Stem conditions indicate that the lowest resultant pressure is **85** psig (which is the auto-start for TBCLC versus NSW).

Technical Reference(s): OP-42, AOP-10 (Attach if not previously provided)

ARP 09-6-2-3

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-46A, EO 1.05.c.2 & 1.14.a/b.1 (As available)

Question Source: [redacted] Bank # [redacted] Modified Bank # [redacted] (Note changes or attach parent)

New X

Question History: [redacted] Last NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 13) 218000 ADS A4.02

Ability to manually operate and/or monitor in the control room:

- ADS logic initiation

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	218000 A4.02	
Importance Rating	4.2	

Proposed Question: **# 97**

A small break LOCA has occurred and no operator action has been taken. HPCI is out of service, RCIC is injecting into the reactor vessel and all low pressure ECCS pumps have started from High Drywell Pressure except the "B" Core Spray Pump which failed to automatically start.

- At Time 0: Reactor water level was 177 inches
- Currently (Time 0 plus one hundred (100) seconds): Reactor water level is 75 inches and dropping.

Which statement correctly describes the operation of the Automatic Depressurization System (ADS)?

a. All 7 ADS valves will open as soon as an additional 34 seconds have elapsed

b. All 7 ADS valves will open 134 seconds after reactor water level drops to 59.5 inches

c. All 7 ADS valves will open 134 seconds after the "B" Core Spray Pump is manually started

d. All 7 ADS valves will open immediately after the "B" Core Spray Pump is manually started

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT – Explanation: The following conditions are required for ADS to automatically open 7 valves. Any low pressure ECCS pump running, reactor low 177 and low-low-low level 59.5 after 134 second timer times out.

a. Incorrect – Have not reached the triple-low setpoint for timer initiation as of the current time.

c. Incorrect – Criteria is **any** low-pressure ECCS Pump running, which has already been met, but still have not reached the triple-low setpoint for timer initiation as of the current time.

d. Incorrect – See above.

Technical Reference(s): ARP 09-4-1-28 (Attach if not previously provided)

TS Table 3.3.5.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J, EO 1.05.c.1, 1.10.b, & 1.14.c (As available)

Question Source: Bank # 21344

Modified Bank # [redacted]

New

(Note changes or attach parent)

Question History: Last NRC Exam LOI-01-01 (11/5/01)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments:

Examination Outline Cross-reference:

(Page 7) 295010 High Drywell Pressure AA1.02

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:

- Drywell floor and equipment drain sumps

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295010 AA1.02	
Importance Rating	3.6	

Proposed Question: **# 99**

The plant is operating at 90% power during a startup, when the following occurs:

At Time 0: Annunciator 09-4-2-23 "RX VESSEL FLANGE SEAL LEAKAGE" alarms.

At Plus Three (3) Minutes:

- CRS notices that Drywell Pressure is 2.6 psig and steadily rising, and
- SNO reports Drywell Floor Drain Sump (FDS) Pump 20P-1B running on a steady, high FDS level

At Plus Eight (8) Minutes:

- SNO reports receipt of annunciator 09-4-1-10 "DW FLOOR SUMP HI-HI OR LO-LO" with neither FDS Pump running, and
- SNO observes Drywell Equipment Drain Sump (EDS) Pump 20P-5A running with EDS level trending up slowly.

Which of the following events is the probable cause of the final operating configuration?

a. The "Alpha" Floor Drain Pump has failed to auto-start on its alternating cycle with "Bravo", and now the FDS overflow to the EDS is exceeding the capacity of the running Equipment Drain Pump.

b. A High Drywell Pressure PCIS Group 2 isolation was generated, but only the "Bravo" side isolation actually occurred, and now the Equipment Drain Pump is running at shut-off head.

c. **A High Drywell Pressure PCIS Group 2 isolation was generated, and a successful isolation resulted in the trip of both Floor Drain Pumps and NO trip of the Equipment Drain Pump.**

d. The "Bravo" Floor Drain Pump has tripped and a logic failure of the associated Drywell Floor Drain Sump Fill Timer has failed to auto-start the "Alpha" Floor Drain Pump.

Proposed Answer: **c.**

Explanation (Optional):

c. CORRECT: Stem conditions of fairly rapid and significant rise in Drywell pressure; still climbing. A PCIS Group 2 Isolation, at 2.7 psig, would auto-close both sump's inboard & outboard isolation valves. On the FDS, either valve closure trips the pump(s). On the EDS, there are no pump trips on valve closure, as the valve cycles versus the pump (always running on recirc).

a. Incorrect: See above. Pumps do alternate, but not in this manner and FDS would not overflow to EDS.

b. Incorrect: See above. Signal generation correct, each sump has an isolation valve on A and B PCIS, EDS Pump on recirc.

d. Incorrect: See above. No pump auto-start logic associated with Fill Timer whatsoever – level alarms only.

Technical Reference(s): OP-50 (Attach if not previously provided)
AOP-15 / ARP-09-4-2-23

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C, EO 1.09.c (As available)
SDLP-20D, EO 1.05.c.1

Question Source: Bank # [redacted]
Modified Bank # 23829 (attached) (Note changes or attach parent)
New [redacted]

Question History: [redacted] NRC Exam [redacted]

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☐
Comprehension or Analysis ☑

10 CFR Part 55 Content: 55.41 ☑
55.43 ☐

Comments:

PLANT	ExamDate	ExamType	QuestionId	NSSSType
Susquehanna 1	8/1/2002	ILO	23829	BWR

A scram has occurred on Unit 1 with drywell pressure above the trip setpoint due to a small leak in the containment. An operator is directed to pump down the Drywell floor drain sump for sampling to determine the leak source. The following events are observed during the evolution:

"Drywell Floor Drain pump (1P402A) control switch was taken to RUN
"Inboard (HV-16108A1) and Outboard (HV-16108A2) Drywell floor drain sump isolation valves where noted to be closed
"Drywell Floor Drain pump tripped after 45 seconds

Which one of the following describes the response of the system under these conditions?

Answer **Response was as expected, isolation valves did not auto open due to a high drywell pressure signal.**

Distractor 1 Response was unexpected, isolation valves should have been open due to a high drywell pressure signal.

Distractor 2 Response was as expected, isolation valves must be opened manually prior to starting sump pump.

Distractor 3 Response was unexpected, isolation valves should have auto opened after pump start.

RefMaterial

QuestionComment d. correct - >1.72 psig isolates valves preventing pump operation for >45 sec. From start signal

Distract1Comment a. incorrect - valves will not auto open with isolation signal in

b. incorrect - valves are normally closed and will not auto open with isolation signal in

c. incorrect - valve auto open if isolation signal is not in

Distract2Comment a. incorrect - valves will not auto open with isolation signal in

b. incorrect - valves are normally closed and will not auto open with isolation signal in

c. incorrect - valve auto open if isolation signal is not in

Distract3Comment a. incorrect - valves will not auto open with isolation signal in

b. incorrect - valves are normally closed and will not auto open with isolation signal in

c. incorrect - valve auto open if isolation signal is not in

KaNumber	KaSegment1	KaSegment2	KaSegment3	KaSegment4	KaSegment5
295010.AA1.02			295010	AA1	02
KaRevision				CognitiveLevel	ExamLevel
				2	R

Examination Outline Cross-reference:

(Page 23) G2.4.2
Knowledge of system setpoints / interlocks and automatic actions associated with EOP entry conditions.

Level	RO
Tier #	3
Group #	[redacted]
K/A #	Cat 4 G2.4.2
Importance Rating	3.9

Proposed Question: **# 100**

A single plant parameter trend has resulted in a transient in which all systems responded as expected. You observe that HPCI has initiated. RCIC remains in standby. Your SNO reports that Reactor Building Ventilation has isolated.

As a result of this information you.....

a. enter EOP-2, RPV Control only.
b. **enter EOP-2, RPV Control, and EOP-4, Primary Containment Control.**
c. enter EOP-5, Secondary Containment Control only.
d. enter EOP-2, RPV Control, and EOP-5, Secondary Containment Control.

Proposed Answer: **b.**

Explanation (Optional):

b. CORRECT: HPCI initiated with RCIC in standby can only result from High Drywell Pressure. High Drywell pressure requires entry into EOP-2 and 4.

a. Incorrect: See above. A low RPV Water Level will result in both HPCI and RCIC initiation and is only an EOP-2 entry.

c. Incorrect: RBV isolates on high drywell pressure but this is not an EOP-5 entry condition. EOP-5 is entered on RBV isolation on high radiation yet this does not result in a HPCI start.

d. Incorrect: See Above. There is no single parameter that will result in both EOP-2 and EOP-5 entry.

Technical Reference(s): EOP-2, 4 and 5 (Attach if not previously provided)

AP-12.03, Step 8.9.3.I

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11C EO-1.02, MIT-301.11 EO-4.02

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ☑

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41 ☑

55.43 ☐

Comments: