Examination Outline Cross-reference:	Level	RO	SRO
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	Tier #	1	
AK1.02 (10CFR 55.41.8 TO 41.10)	Group #	1	
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :	K/A #	295001AK1.02	
• Power/flow distribution	Importance Rating	3.3	

Proposed Question: **# 1**

Unit 1 is at 100% Reactor Power **AND** Core Flow is 92%. A trip of 1A Recirc Pump results in Operation in Region II of the Core Power to Flow Map.

Which ONE of the following completes the statement below?

The required action(s) in accordance with 1-AOI-68-1A, "Recirc Pump Trip / Core Flow Decrease," is (are) to **IMMEDIATELY** ________.

A. insert a Manual Reactor Scram

B. raise Core Flow until Region II of the Power to Flow Map is exited

C. insert Control Rods until Region II of the Power to Flow Map is exited

D. insert Control Rods until Load Line is < 95.2%; then, raise Core Flow to > 45%

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that IF both Recirc Pumps are tripped in Modes 1 or 2, THEN 1-AOI-68-1A requires the Reactor to be Scrammed.

B INCORRECT: Plausible in that immediately raising core flow would be an expeditious method to exit instability regions. If load line was less than 95.2% following the Recirc Pump trip, this would be the correct answer.

C INCORRECT: Plausible in that Control Rod are required to be immediately inserted if in Region I or II but the crew will stop inserting Control Rods when Load Line is < 95.2%. That is, Control Rod insertion will stop prior to exiting the Region and raising core flow will complete the exit from Region II. If core flow was greater than 45% following the Recirc Pump Trip, this would be the correct answer.

D **CORRECT:** In accordance with 1-AOI-68-1A, IF Region I or II of the Power to Flow Map is entered due to a trip of a Recirc Pump, THEN IMMEDIATELY take actions to insert control rods to less than 95.2% loadline. Then, RAISE core flow to greater than 45% in accordance with1-OI-68.

KA Justification:

The KA is met because it tests candidate's knowledge of operational implications of Reactor Power / Flow distribution with a partial loss of core circulation as a result of a Recirc Pump trip.

Question Cognitive Level:

Question rated as C/A because Candidates' must process multiple pieces of data to determine correct actions in accordance with 1-AOI-68-1A. Candidate must recognize that with core flow of 92% at Reactor Power of 100% that Load Line is greater 100% and will remain greater than 100% following the Recirc Pump trip. Also, must recognize that following the trip, Core Flow will be less than 45% requiring increase in core flow also.

Technical Reference(s): 1-AOI-68-1A Rev 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.007 V.B.28 (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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295003 Partial or Complete Loss of A.C. Power / 6	Tier #	1	------
G2.4.6 (10CFR 55.41.10)	Group #	1	------
Knowledge of EOP mitigation strategies.	K/A #	295003 G2.4.6	
	Importance Rating	3.7	------

Proposed Question: **# 2**

A leak in the Unit 1 Drywell results in the following conditions:

- Drywell Temperature is 170° F and rising
- A Lockout occurs on 4kV Shutdown Board C
- Reactor Level is (+) 10 inches and stable
- Suppression Pool Level is 15 feet

Which ONE of the following completes the statements below?

In accordance with 1-EOI-2, "Primary Containment Control," Drywell Spray must be initiated before MAXIMUM Drywell Temperature of __(1)__ . Assuming no manual electric board transfers are performed, RHR __(2)__ is (are) available for Drywell Spray from the control room.

A. **(1)** 200° F
(2) Loop I **ONLY**

B. **(1)** 200° F
(2) Loop I **AND** Loop II

C. **(1)** 280° F
(2) Loop I **ONLY**

D. **(1)** 280° F
(2) Loop I **AND** Loop II

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

B INCORRECT: Part 1 incorrect – Plausible in that Drywell Temperature of 200° F is a recognizable value of 1-EOI-2, Drywell Temp Leg requiring entry into EOI-1. Part 2 incorrect – Plausible in that Unit 2 480 V Shutdown Board B is supplied from 4 kV S/D Board D. On Unit 2 this would be the correct answer.

C **CORRECT**: Part 1 correct – 1-EOI-2 directs Drywell Spray prior to Drywell Temp of 280° F. Part 2 correct – Loop II Drywell Spray valves are powered from 480 RMOV Board B which is powered from 480 V S/D Board B. This Board is powered from 4 kV S/D Board C on Unit 1 which is locked out. Although one pump is available on Loop 2, Spray Valves can not be opened from the control room.

D INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

KA Justification:

The KA is met because question tests knowledge of EOI mitigation strategies with partial loss of AC Power.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidate must determine effect of a Lockout on 4kV Shutdown Board C on ability to Spray the Drywell.

Technical Reference(s): OPL171.036 Rev. 12 / 1-EOI-2 Rev. 1 (Attach if not previously provided)
OPL171.044 Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 V.B.19 (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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295004 Partial or Total Loss of DC Pwr / 6	Tier #	1	------
AA1.03 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:	K/A #	295004AA1.03	
• A.C. electrical distribution	Importance Rating	3.4	------

Proposed Question: **# 3**

Unit 2 was operating at 100% Reactor Power.

A ground **AND** subsequent fire in Shutdown Board 250V DC Distribution Panel SB-B resulted in de-energization of the SB-B panel **AND** trip of 4kV Shutdown Board B Normal Feeder Breaker.

Which ONE of the following completes the statements below?

480V Shutdown Board 2B is __**(1)**__.

4kV Shutdown Board B __**(2)**__ automatically transfer to its alternate source.

A. **(1)** energized
(2) will

B. **(1)** de-energized
(2) will

C. **(1)** energized
(2) will **NOT**

D. **(1)** de-energized
(2) will **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See explanation C. Part 2 incorrect – See explanation B.

B INCORRECT: Part 1 incorrect - 480v Shutdown Board 2B remains energized with the loss of 4kV Shutdown Board B. Plausibility based on misconception 480v Shutdown Board B normal power supply would be from 4kV Shutdown Board B. If this was Unit 1 480 V and 4Kv A Shutdown Boards, this would be the correct answer. Part 2 incorrect - Each Shutdown Battery system supplies its respective 4KV Shutdown Board and 480V Shutdown Board. All control power transfers are manual. Plausible in that if control power transfer is automatic as board power supply is or control power was not from SB-B DC Distribution Panel, this would be the correct answer

C CORRECT: Part 1 correct - 480v Shutdown Board 2B remains energized with the loss of 4kV Shutdown Board B. 4kV Shutdown Board D is the normal feeder to the 480v S/D Bd 2B. Part 2 correct - Each Shutdown Battery system supplies its respective 4KV Shutdown Board and 480V Shutdown Board. All control power transfers are manual. With the loss of control power, normal automatic transfer to alternate power supply will not occur.

D INCORRECT: Part 1 incorrect – See explanation A. Part 2 correct – See explanation D.

KA Justification:

The KA is met because to successfully answer this question, candidate must recognize the impact of partial loss of DC (SB-B Distribution Panel) will have on control power to 4 kV Shutdown Board B and the impact of loss of 4kV Shutdown Board B will have on 480v Shutdown Board 2B.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.036 Rev 12 (Attach if not previously provided)
OPL171.037 Rev 12
0-OI-57B Rev 189

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 V.B.1 (As available)
OPL171.036 V.B.6/8

Question Source: Bank #
Modified Bank # BFN 1006 #3 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 2010

(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 **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295005 Main Turbine Generator Trip / 3	Tier #	1	------
AK1.01 (10CFR 55.41.8 to 41.10)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP :	K/A #	295005AK1.01	
• Pressure effects on reactor power	Importance Rating	4.0	------

Proposed Question: **# 4**

Given the following conditions:

- Unit 3 is operating at 20% Reactor Power
- A pipe rupture results in loss of **ALL** EHC with the Main Turbine online

Which ONE of the following completes the statements below?

Reactor Pressure will __(1)__ .

An automatic scram __(2)__ occur.

A. **(1)** rise
(2) will

B. **(1)** lower
(2) will

C. **(1)** rise
(2) will **NOT**

D. **(1)** lower
(2) will **NOT**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** With the failure of EHC, the Main Turbine Trips and Bypass Valves will fail closed. Reactor Pressure will rise until the Reactor High Pressure Scram setpoint is reached.

B INCORRECT: Plausibility based on misconception that Bypass Valves fail open on loss of EHC and subsequent scram on MSIV closure. Failing open is plausible in that there are EHC failures which will result in Bypass Valves failing open. For example, with EHC Control System in HEADER PRESSURE CONTROL, a single Header Pressure input failing high would result in Main Turbine Control Valves and Bypass Valves opening in attempt lower Reactor Pressure. Additionally, 3-AOI-47-2, "Turbine EHC Control System Malfunctions," addresses EHC System Failures which result in lowering Reactor Pressure.

C INCORRECT: Plausible in that if candidate considers only Main Turbine Trip actuation of RPS, this would be the correct answer since it is bypassed at this power level.

D INCORRECT: Plausibility based on misconceptions that Bypass Valves fail open on loss of EHC as discussed in detail above and subsequent scram on MSIV closure is bypassed at this power level or candidate considers only Main Turbine Trip actuation of RPS.

KA Justification:

The KA is met because the question tests knowledge of the operational implications of Pressure effects on reactor power as they apply to Main Turbine Generator Trip due to loss of EHC.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.010, Rev. 12 (Attach if not previously provided)

3-OI-99 Rev. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010, V.B.6 (As available)

OPL171.010, V.B.23

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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

295006 SCRAM / 1

AA1.05 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to SCRAM :

- Neutron monitoring system

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295006AA1.05	
Importance Rating	4.2	------

Proposed Question: **# 5**

Given the following:

- Unit 2 in Mode 2
- Intermediate Range Monitors (IRMs) indicate 29.1 on Range 3
- Reactor Period is 90 seconds.

Which ONE of the following identifies approximately how long it will take to reach the IRM Scram setpoint?

A. 35 seconds

B. 65 seconds

C. 125 seconds

D. 180 seconds

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that this would be half the time to the first doubling.

B INCORRECT: Plausible in that this would be the time to the first doubling.

C **CORRECT**: C is correct as with a reactor period of 90 and 2 doubling times, (29.1-58.2 and 58.2-116.4). This time would be 62.28 seconds times 2. The scram setpoint would be reached in 124.56 seconds.

D INCORRECT: Plausible in that this would be twice the period.

KA Justification:

The KA is met because the question tests candidates' ability to monitor IRMs as they apply to Scram.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidates must determine doubling time based on Reactor Period then calculate time to reach IRM Scram setpoint.

Technical Reference(s):	OPL171.020, Rev. 11 / 2-OI-92A, Rev. 28 2-GOI-100-1A Rev. 145	(Attach if not previously provided)
Proposed references to be provided to applicants during examination:	NONE	
Learning Objective:	OPL171.020 V.B.7	(As available)
Question Source:	Bank #	Monticello 07 #43
	Modified Bank #	(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	Monticello 2007

(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	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295016 Control Room Abandonment / 7	Tier #	1	------
G2.1.28 (10CFR 55.41.7)	Group #	1	------
Knowledge of the purpose and function of major system components and controls.	K/A #	295016G2.1.28	
	Importance Rating	4.1	------

Proposed Question: **# 6**

Which ONE of the following functions can be performed at Backup Control Panel 2-25-32?

A. Close **ALL** MSIVs

B. Operate **ALL** ADS Valves

C. Suppression Chamber Spray

D. Control Reactor Level with HPCI

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: BOTH Inboard and Outboard MSIVs can be closed from Backup Control Panel 2-25-32.

B INCORRECT: Plausible in that Four ADS valves can be controlled from Panel 25-32. Six SRVs (Non-ADS) have disconnect switches at Panel 25-32.

C INCORRECT: Plausible in that indications for RHR are on 2-25-32 and 2-AOI-100-2, "Control Room Abandonment," provides instruction for Suppression Pool Cooling and Shutdown Cooling.

D INCORRECT: Plausible in that Reactor Level can be controlled with RCIC at Pnl 2-25-32.

KA Justification:

The KA is met because it tests the candidate's knowledge of function of major system components associated with Control Room Abandonment procedure and the Backup Control Panel.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s):	2-AOI-100-2, Rev. 54	(Attach if not previously provided)
	OPL171.208, Rev. 5	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (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	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295018 Partial or Complete Loss of Component Cooling Water / 8	Tier #	1	------
AA2.01 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :	K/A #	295018 AA2.01	
• Component temperatures	Importance Rating	3.3	------

Proposed Question: **# 7**

Unit 3 is operating at 100% Reactor Power when the following alarms **AND** indications are received:

- A Partial Loss of Reactor Building Closed Loop Cooling Water occurs.
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH, (3-9-4B, Window 17) is in alarm.
- RWCU Non- Regenerative Heat Exchanger Discharge Temperature is 140° F.

Which ONE of the following completes the statement below?

The Reactor Water Cleanup (RWCU) Pumps will receive a **DIRECT** TRIP input from _____.

A. either the isolation valve position or the non-Regenerative Heat Exchanger high temperature signal

B. either the isolation valve position or a low flow condition after 30 seconds

C. the non-Regenerative Heat Exchanger high temperature signal **ONLY**

D. a low flow condition after 30 seconds **ONLY**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** Plausible in that this is a misconception about RWCU Pump Trip directly from High Temperature signal. High Temperature initiates a PCIS Isolation. When the Isolation Valve is NOT FULLY OPEN, the RWCU Pump TRIPS, therefore the first part of the statement is correct and the second part is incorrect.

B **CORRECT**: RWCU Non- Regenerative Heat Exchanger Discharge Temperature at 140° F isolates RWCU. When RWCU isolation valve FCV 69-1or 2 Not Full Open, RWCU Pumps trip. If pumps did not trip on FCV 69-1 or 2 Not Full Open, and the valves had traveled closed, then pumps would trip on low flow with a 30 second time delay.

C INCORRECT: Plausible in that identifies misconception about RWCU Pump Trip directly from High Temperature signal. High Temperature initiates a PCIS Isolation.

D INCORRECT: Plausible in that System Low Flow of 56 gpm with a time delay of 30 seconds will trip RWCU Pumps. With the Isolation Trip coming with valves just off full open, they would cause the trip prior to low flow condition.

KA Justification:

This question satisfies the K/A statement by testing candidates' ability to interpret RWCU Temperatures as they apply to Partial Loss of RBCCW. Partial loss of RBCCW results in RWCU Non- Regenerative Heat Exchanger Discharge Temperature at 140° F which isolates RWCU. When RWCU isolation valve FCV 69-1or 2 Not Full Open, RWCU Pumps trip.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

Technical Reference(s): 3-OI-69 Rev. 79 (Attach if not previously provided)

OPL171.013 Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.013 V.B.3 (As available)

Question Source:

Bank #		
Modified Bank #	Nine Mile 2 08 #45	(Note changes or attach parent)
New		

Question History: Last NRC Exam Nine Mile 2 2008

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295019 Partial or Total Loss of Inst. Air / 8	Tier #	1	------
AK3.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : • Service air isolations: Plant-Specific	K/A #	295109AK3.03	
	Importance Rating	3.2	------

Proposed Question: **# 8**

Control Air Header Pressure is lowering due to a rupture in the system.

Which ONE of the following identifies the **HIGHEST** Control Air Pressure that will result in Service Air Isolation Valve, 0-FCV-33-1, closing **AND** the reason?

A. 30 psig;
To isolate non-essential Service Air loads.

B. 30 psig;
Due to insufficient air pressure to keep the valve open.

C. 50 psig;
To isolate non-essential Service Air loads.

D. 50 psig;
Due to insufficient air pressure to keep the valve open.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: 1st part correct – See B Explanation. 2nd Part incorrect – See C Explanation.

B **CORRECT:** Service air supply valve from control air header (0-FCV-33-1). The valve automatically opens if control air pressure falls to 85 psig and closes at 30 psig (due to insufficient air pressure to keep the valve open).

C INCORRECT: Recognizable pressure associated with loss of Control Air as the pressure that Condensate Demin Bypass Valve Fails open. Plausible in that it is logical to isolate non-essential Service Air loads with a loss of Control Air similar to RBCCW Sectionalizing Valve closing on low header pressure to isolate non-essential RBCCW loads.

D INCORRECT: 1st part incorrect – see C Explanation. 2nd Part Correct – See B Explanation.

KA Justification:

This question satisfies the K/A statement by testing knowledge of the reason and the setpoint for Service air isolation Valve, 0-FCV-33-1, closing as a result of a rupture in the Control Air System and lowering pressure.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 0-OI-32 Rev 127 (Attach if not previously provided)
OPL171.054 Rev 15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.054 V.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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295021 Loss of Shutdown Cooling / 4	Tier #	1	------
AK2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:	K/A #	295021AK2.01	
• Reactor water temperature	Importance Rating	3.6	------

Proposed Question: **# 9**

Unit 3 is in Mode 4 with the following conditions:

- Reactor Level band is (+) 78 inches to support testing.
- **ALL** Reactor Recirc **AND** RWCU Pumps are isolated and tagged out.
- RHR Loop I in Shutdown Cooling experiences an inadvertent Group 2 Isolation **AND** can **NOT** be restored.

Which ONE of the following completes the statements below?

Accurate Reactor Water Temperature __**(1)**__ available.

If Reactor Coolant Stratification occurs, it is indicated by __**(2)**__ .

A. **(1)** is
(2) a MINIMUM differential temperature of 50°F or greater between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

B. **(1)** is **NOT**
(2) a MINIMUM differential temperature of 50°F or greater between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

C. **(1)** is
(2) a MINIMUM differential temperature of 75°F or greater between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

D. **(1)** is **NOT**
(2) a MINIMUM differential temperature of 75°F or greater between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1incorrect - plausible in that Reactor Level is high enough to establish natural circulation. Candidate may believe natural circulation is adequate to provide accurate level indication. Part 2 correct - see B below.

B **CORRECT**: Part 1 correct – In accordance with "Loss of Shutdown Cooling," 3-AOI-74-1, accurate coolant temperatures will not be available if forced circulation is lost. Part 2 correct –in accordance with "Loss of Shutdown Cooling," 3-AOI-74-1, with the Reactor in Cold Shutdown Condition (Mode 4 or Mode 5) coolant stratification may be indicated by Differential temperatures of ≥ 50°F between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

C INCORRECT: Part 1 incorrect – see A above. Part 2 incorrect – Plausible in that in accordance with 3-AOI-68-1A, "Recirc Pump Trip/Core Flow Decrease OPRMs Operable", the temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other, the candidate may mistake this number for the 50°F in 3-AOI-74-1

D INCORRECT: Part 1 correct – See B above. Part 2 incorrect – see C above

KA Justification:

The KA is met because to successfully answer the question, the candidate must demonstrate knowledge of the interrelationship between loss of shutdown cooling and Reactor Water Temp.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

Technical Reference(s): OPL171.074 Rev 8 (Attach if not previously provided)
3-AOI-74-1 Rev 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 V.B.6 (As available)

Question Source: Bank #
Modified Bank # BFN 1006 #9 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295023 Refueling Acc / 8	Tier #	1	------
AA1.03 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: • Fuel handling equipment	K/A #	295023AA1.03	
	Importance Rating	3.3	------

Proposed Question: **# 10**

Unit 1 is in a Refueling Outage. The Refueling SRO reports that an **IRRADIATED** fuel assembly has been seated in the **WRONG** location in the core. The grapple remains engaged on the bundle.

The following conditions are then noted:

- Rising count rates on SRMs
- SRM Period lights illuminated
- Rising dose rates on the Refuel Floor

Which ONE of the following describes an **IMMEDIATE** Operator action in accordance with Refueling AOIs?

A. Verify Secondary Containment is intact.

B. Raise the fuel bundle from the core location.

C. If any CRD Pump is in service stop the CRD Pump.

D. If SLC is operable place SLC PUMP 1A/1B, 1-HS-63-6A control switch in START A **OR** START B.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is plausible because it is a required subsequent action of 1-AOI-79-1, "Fuel Damage During Refueling."

C INCORRECT: This is plausible because it is a required subsequent action of 1-AOI-79-2, not immediate action.

B **CORRECT:** In order to answer this question correctly the candidate must determine the appropriate condition and Immediate Action required by 1-AOI-79-2.

D INCORRECT: This is plausible because it is a required subsequent action of 1-AOI-79-2, not immediate action.

KA Justification:

This question satisfies the K/A statement by requiring the candidate to analyze specific plant conditions to determine appropriate actions to take with fuel handling equipment in response to inadvertent criticality.

Question Cognitive Level:

Fundamental Knowledge

Technical Reference(s): 1-AOI-79-2 Rev. 0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.060 V.B.3 (As available)

Question Source: Bank # BFN 1006 #10

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 2010

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295024 High Drywell Pressure / 5	Tier #	1	------
EK3.08 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : • Containment spray: Plant-Specific.	K/A #	295024EK3.08	
	Importance Rating	3.7	------

Proposed Question: **# 11**

Unit 2 was at 100% Reactor Power when a spurious Group I Isolation occurred. The pressure transient caused a small-break LOCA to occur inside the Drywell.

Which ONE of the following describes the basis for actions with respect to 12 psig Suppression Chamber Pressure?

A. Drywell sprays must be initiated prior to this pressure to prevent opening the Suppression Chamber to Reactor Building vacuum breakers **AND** de-inerting the containment.

B. Drywell sprays must be initiated above this pressure because almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus **AND** chugging is possible.

C. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the torus have been transferred to the drywell air space **AND** Suppression Chamber Sprays will be ineffective.

D. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus so initiating Drywell Sprays may result in containment failure.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is plausible because initiation of DW sprays at high SC pressure could reduce pressure low enough to open the Suppression Chamber to Reactor Building Vacuum Breakers. However, this is part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

B **CORRECT**: Drywell sprays must be initiated above this pressure because almost all of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus **AND** chugging is possible. The basis for the Pressure Suppression Pressure Limit of 12 psig Suppression Chamber pressure.

C INCORRECT: This is plausible if the LOCA occurred inside the Suppression Chamber and NOT the Drywell as given in the stem.

D INCORRECT: This is plausible because initiating SC sprays with high temperature non-condensable gases in the SC will result in evaporative cooling and a rapid pressure drop. However, the SC to DW vacuum relief system is capable of compensating for this pressure drop. This is also part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

KA Justification:

The KA is met because it tests knowledge of the reasons for Drywell Spray as it applies to High Drywell Pressure.

Question Cognitive Level:

This question is rated as Memory due to the requirement to recall or recognize discrete bits of information.

Technical Reference(s): EOIPM Section 0-V-D Rev. 0 (Attach if not previously provided)
OPL171.203 Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 V.B.5 (As available)

Question Source: Bank # BFN 0610 #62
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 2008

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295025 High Reactor Pressure / 3	Tier #	1	------
EA1.04 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: • HPCI: Plant-Specific	K/A #	295025EA1.04	
	Importance Rating	3.8	------

Proposed Question: **# 12**

Unit 1 HPCI is in operation in Pressure Control Mode per 1-EOI Appendix 11C, "ALTERNATE RPV PRESSURE CONTROL SYSTEMS HPCI TEST MODE."

- Reactor Pressure is 1050 psig
- 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, is in Automatic

Which ONE of the following completes the statement below?

To lower Reactor Pressure, the operator is required to use __**(1)**__ **AND** __**(2)**__ in accordance with 1-EOI Appendix 11C.

A. **(1)** 1-FCV-73-36, HPCI/RCIC CST TEST VLV,
(2) throttle it in the CLOSE direction

B. **(1)** 1-FCV-73-36, HPCI/RCIC CST TEST VLV,
(2) throttle it in the OPEN direction

C. **(1)** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL,
(2) LOWER the setpoint

D. **(1)** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL,
(2) RAISE the setpoint

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that 1-FCV-73-35, HPCI PUMP CST TEST VLV is adjusted in accordance with 1-EOI Appendix 11C to control HPCI pump discharge pressure at or below 1100 psig.

B INCORRECT: See Explanation A.

C INCORRECT: Second Part is incorrect – Plausibility based on misconception that lowering setpoint will result in lowering Reactor Pressure.

D **CORRECT:** Both parts are correct – Per 1-EOI Appendix 11C, ADJUST 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller to control RPV pressure. Raising set point will lower reactor pressure, per the appendix..

KA Justification:

The KA is met because the question tests the candidates' ability to operate and monitor HPCI in pressure control mode as it applies to high Reactor Pressure.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-EOI Appendix 11C Rev. 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 V.B.10 (As available)

Question Source: Bank # Hatch 09 #52

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Hatch 2009

(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 **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295026 Suppression Pool High Water Temp. / 5	Tier #	1	------
EK2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following:	K/A #	295026EK2.02	
• Suppression pool spray: Plant-Specific	Importance Rating	3.6	------

Proposed Question: **# 13**

Unit ~~3~~ 2 has experienced a LOCA **AND** the following conditions exist:

- Suppression Chamber Pressure is 5 psig
- Suppression Pool level is 14.5 feet
- Drywell Pressure is 7.5 psig
- Suppression Pool Temperature is 205° F
- **BOTH** RHR Loop I Pumps are in Suppression Chamber / Drywell Spray with Loop flow of 11,500 gpm
- Core Spray Pump 2A flow is 4000 gpm
- RHR Pump 2B flow is 10500 gpm
- **NO** other ECCS Pumps are running

Based on the above conditions, which ONE of the following identifies the ECCS Pump(s) that has (have) sufficient NPSH for continued operation?

[REFERENCE PROVIDED]

A. Core Spray Pump 2A **ONLY**

B. RHR Pumps 2A **AND** 2C **ONLY**

C. RHR Pumps 2A, 2B **AND** 2C **ONLY**

D. Core Spray Pump 2A **AND** RHR Loop I Pumps

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Core Spray Pump 2A above the safe region of NPSH Limits Curve 1. Plausible in that if Drywell pressure is used to plot Curve 1, Pump would be operating in the safe region of curve 1 and if RHR is Plotted for Loop flow, it would be in the Unsafe of Curve 2.

B **CORRECT:** Operating point for RHR Loop I Pumps is within the safe region of Curve 2.

C INCORRECT: Plausible in that If RHR B flow is plotted as loop flow for two pumps in operation, this would be the correct answer.

D INCORRECT: RHR Loop I Pumps have adequate NPSH. However, CS Pump 2A does not. Plausible in that if Drywell pressure is used to plot both Curves, all Pumps would be operating in the safe regions and this would be the correct answer.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the interrelationship between High Suppression Pool Temperature and RHR Spray Operation.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question and use a reference to solve a problem.

Technical Reference(s): 3-EOI-1 Curve 1 / Curve 2 Rev. 8 (Attach if not previously provided)

OPL171.201 Rev. 7

Proposed references to be provided to applicants during examination: CS NPSH Limit Curve 1

RHR NPSH Limit Curve 2

Learning Objective: OPL171.201 V.B.13 (As available)

Question Source: Bank #

Modified Bank # BFN 1006 #15 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295028 High Drywell Temperature / 5	Tier #	1	------
EK1.01 (10CFR 55.41.8)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:	K/A #	295028EK1.01	
• Reactor water level measurement	Importance Rating	3.5	------

Proposed Question: **# 14**

Given the following Unit 2 plant conditions:

- Reactor pressure is being maintained at 50 psig
- Temperature near the water level instrument run in the Drywell is 220° F
- The Shutdown Vessel Flooding Range Instrument (2-LI-3-55) is reading (+) 35 inches

Which ONE of the following is the **HIGHEST** Drywell Run Temperature at which the 2-LI-3-55 reading (+) 35 inches is considered valid?

[REFERENCE PROVIDED]

A. 200° F

B. 250° F

C. 270° F

D. 300° F

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is plausible since 200°F is a valid indication; however the question calls for the HIGHEST temperature.

B **CORRECT**: In order to answer this question correctly, the candidate must use EOI Caution #1 to determine operable RPV water level instruments.

C INCORRECT: This is plausible if the candidate interpolates the Caution #1 table, however this is NOT permissible.

D INCORRECT: This is plausible if the candidate uses only Curve 8.

KA Justification:

The KA is met because it tests knowledge of the operational implications of Reactor water level measurement with High Drywell Temperature near the water level instruments runs.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

Technical Reference(s): OPL171.201 Rev 7 (Attach if not previously provided)
2-EOI-1 Rev 12 (Including version / revision number)

Proposed references to be provided to applicants during examination: **2-EOI Caution #1 and Curve 8**

Learning Objective: OPL171.201 V.B.13 (As available)

Question Source: Bank # BFN 0610 #73
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 0610

(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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295030 Low Suppression Pool Wtr Lvl / 5	Tier #	1	
G2.1.31 (10CFR 55.41.10)	Group #	1	
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	K/A #	295030G2.1.31	
	Importance Rating	4.6	

Proposed Question: **# 15**

Unit 3 was at 100% Reactor Power when a leak from the Torus resulted in Suppression Pool Level of 11.4 feet. Required actions of the EOIs have been performed.

Which ONE of the following completes the statement below?

Two minutes after initiating required EOI actions, Wide Range Reactor Pressure Indication(s) available on Control Room Panel(s) __(1)__ will be __(2)__.

A. **(1)** 3-9-5 **ONLY**
(2) stable

B. **(1)** 3-9-5 **ONLY**
(2) lowering

C. **(1)** 3-9-3 **AND** 3-9-5
(2) stable

D. **(1)** 3-9-3 **AND** 3-9-5
(2) lowering

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that this would be the correct answer if the question asked where Narrow Range Pressure indication is available. Part 2 incorrect – Plausible in that in accordance with 3-EOI-2, reactor scram is required if Suppression Pool can not be maintained >11.5 feet. Two minutes after the scram, reactor pressure would be stable. However, this is incorrect since 3-EOI-2 also required ED for this condition.

B INCORRECT**:** Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT**: Part 1 correct – Wide Range Pressure indication is available on both 3-9-3 and 3-9-5. Part 2 correct – Per 3-EOI-2, if Suppression Pool Level can not be maintained > 11.5 feet, Reactor Scram and Emergency Depressurization are required.

KA Justification:

The KA is met because the question tests candidates' ability to locate control room wide range pressure indications, and to determine that they correctly reflect the desired plant lineup which is lowering pressure due to requirement to ED on Low Suppression Pool Level.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

Technical Reference(s): 3-EOI-1 Rev. 8 / 3-EOI-2 Rev. 8 (Attach if not previously provided)

OPL171.003 Rev. 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 V.B.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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

295031 Reactor Low Water Level

K3.01 (CFR 41.5)

Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL :

- Automatic depressurization system actuation

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295031K3.01	
Importance Rating	3.9	------

Proposed Question: **# 16**

Given the following Unit 1 plant conditions:

- HPCI 120VAC POWER FAILURE, (1-9-3F, Window 7) is in alarm
- Reactor Water Level is (-) 122 inches and lowering
- Drywell Pressure is 1.8 psig and steady
- Assume **NO** operator action

Which ONE of the following completes the statements below?

ADS will automatically initiate in __**(1)**__ . This actuation is in response to a LOCA __**(2)**__.

A. **(1)** 265 seconds
(2) inside the Drywell

B. **(1)** 360 seconds
(2) inside the Drywell

C. **(1)** 265 seconds
(2) outside the Drywell

D. **(1)** 360 seconds
(2) outside the Drywell

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - This time delay is associated with -122 inches received without a high DW pressure (>2.45 psig), which is given in the stem. However, once this timer times out, if ECCS pumps are running, a 95 second timer initiates and must time out before ADS initiates. This makes the total time 360 seconds. Part 2 incorrect - This is the basis for ADS initiation with BOTH high DW pressure AND low RPV level.

B INCORRECT: Part correct as stated in D. Part 2 incorrect as stated in A above.

C INCORRECT: Part 1 incorrect as stated in A above. Part 2 correct. ADS initiation in the absence of high DW pressure is due to decay heat boil-off following a LOCA outside the Drywell with MSIV isolation.

D **CORRECT:** Part 1 correct - Time delay associated with -122 inches received without a high DW pressure >2.45 psig (265 sec), plus the 95 second timer makes the total time 360 seconds. Part 2 correct. ADS initiation in the absence of high DW pressure is due to decay heat boil-off following a LOCA outside the Drywell with MSIV isolation.

KA Justification:

The KA is met because the question tests knowledge of the reason for Automatic Depressurization system actuation as it applies to Low Reactor Water Level.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

Technical Reference(s): OPL171.043 Rev. 13 (Attach if not previously provided)

1-OI-1 Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.043 V.B.4 (As available)

Question Source: Bank # BFN 0707 #54

Modified Bank #

New

(Note changes or attach parent)

Question History: Last NRC Exam Browns Ferry 2007

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: The addition of "Assume NO operator action" was added due to procedural guidance which would inhibit ADS initiation under this condition. In this condition, 1-EOI-1 flowchart path RC/L would allow ADS to be inhibited below -100 inches. In addition, 1-EOI-C1 would be entered below approximately -120 inches and direct that ADS be inhibited. In fact, there are no foreseeable circumstances where ADS would be allowed to auto initiate by procedure.

The HPCI 120VAC Power Failure annunciator is to provide realistic conditions where ADS would auto initiate. If HPCI were operable, ADS would not be required under these conditions.

Examination Outline Cross-reference:	Level	RO	SRO
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	Tier #	1	------
EA2.06 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :	K/A #	295037EA2.06	
• Reactor pressure	Importance Rating	4.0	------

Proposed Question: **# 17**

An ATWS has occurred on Unit 1 with the following time line **AND** conditions:

- At 1200 Reactor Power is 15%
- At 1210 SLC is initiated
- At 1235 SLC Storage Tank Level is 67%
- At 1300 SLC Storage Tank Level is 43%

Which ONE of the following completes the statements below?

In accordance with 1-EOI-1, "RPV Control," __**(1)**__ is the earliest time the crew must commence depressurizing the Reactor below the Shutdown Cooling Reactor Pressure interlock.

Cooldown rate of 100° F/hour __**(2)**__ be exceeded.

A. **(1)** 1235
(2) can

B. **(1)** 1235
(2) **CANNOT**

C. **(1)** 1300
(2) can

D. **(1)** 1300
(2) **CANNOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Level must be 43% to commence cooldown. Plausible in that 67% tank level is Hot Shutdown weight for SLC. Part 2 incorrect – Plausible in that under certain conditions in EOI-1, cooldown is performed irrespective of cooldown rates.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT**: Part 1 correct – In accordance with 1-EOI-1, when SLC has been injected into the RPV to a tank level of 43%, depressurize the RPV below the shutdown cooling pressure interlock. Part 2 correct – Must maintain cooldown rate < 100° F per hour.

KA Justification:

The KA is met because the question tests the candidates' ability to determine when Reactor Pressure is lowered in accordance with the EOIs with an ATWS condition present.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome

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Technical Reference(s): 1-EOI-1, Rev. 0 (Attach if not previously provided)

OPL171.202 Rev. 8

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.039, V.B.6 (As available)

OPL171.202, V.B.9

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

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-Site Release Rate	Tier #	1	------
EK2.10 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following:	K/A #	295038EK2.10	
• Condenser air removal system	Importance Rating	3.2	------

Proposed Question: **# 18**

Unit 2 is in Start Up. Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS. The following alarms/indications are received:

- OG POST-TREATMENT RADIATION HIGH, (2-9-4C, Window 33)
- Offgas Post-Treatment Radiation is 6.5×10^4 cps

Which ONE of the following identifies the impact of this condition on the Offgas System?

A. **NO** valves will reposition

B. Adsorber Bypass Valve, 2-FCV-66-113B will close. **NO** other valves will reposition.

C. Adsorber Bypass Valve, 2-FCV-66-113B will close **AND** Adsorber Inlet Valve, 2-FCV-66-113A will open. **NO** other valves will reposition.

D. Adsorber Bypass Valve, 2-FCV-66-113B will close. Adsorber Inlet Valve, 2-FCV-66-113A **AND** Charcoal Adsorber Train 2 Inlet Valve, 2-FCV-66-118 will open.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: With Off Gas Treatment Select Switch, 2-XS-66-113, not in AUTO, the Radiation High will not result in automatic alignment of Offgas Charcoal Adsorbers.

B INCORRECT**:** Plausibility based on misconception that only Adsorber Bypass Valve, 2-FCV-66-113B will close on High Radiation and that the function remains in force with the Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS.

C INCORRECT**:** If Off Gas Treatment Select Switch, 2-XS-66-113, was in AUTO, this would be the correct answer. Adsorber Bypass Valve (FCV-66-113B) will close, and Adsorber Inlet Valve (FCV-66-113A) will open when one channel reaches OG POST-TREATMENT RADIATION HIGH. Plausible in that the 3 X High Radiation Offgas isolation will occur with the Off Gas Treatment Select Switch, 2-XS-66-113 in any position.

D INCORRECT: Plausibility based on misconception that Charcoal Adsorber Train 2 Inlet Valve, 2-FCV-66-118 will open on High Radiation and that the function remains in force with the Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS. Plausible in that when aligning charcoal filters for parallel operation, 2-OI-66 directs opening of this valve.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between High Off-Site Release Rate as indicated by Offgas Post Treat Radiation High and the Condenser air removal system including the response of Adsorber Bypass Valve, FCV-66-113B, **AND** the Adsorber Inlet Valve, FCV-66-113A. Since there is no procedural guidance for operation with the Off Gas Treatment Select Switch, 2-XS-66-113, in AUTO in any conditions, the question is posed with the Select Switch in BYPASS for operational validity.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.033 Rev. 13 (Attach if not previously provided)
OPL171.030 Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 V.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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
600000 Plant Fire On Site / 8	Tier #	1	------
AA2.13 (10CFR 55.41.10)	Group #	1	------
Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE:	K/A #	600000AA2.13	
• Need for emergency plant shutdown	Importance Rating	3.2	------

Proposed Question: **# 19**

With **ALL** 3 Units operating at 100% Reactor Power, a fire at 4 kV Shutdown Board A has resulted in the following:

- Failure of Unit 1 RHR Pump 1A **AND** Core Spray Pump 1A
- Shift Manager has declared an Appendix R Fire

In accordance with Safe Shutdown Instructions, which ONE of the following identifies which, if any, Reactor(s) is (are) required to be scrammed?

A. **NO** Reactor Scram is required

B. Unit 1 **ONLY**

C. Unit 1 **AND** Unit 2 **ONLY**

D. **ALL** 3 Units

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that no conditions have been identified which would require a Reactor Scram in accordance with AOIs (including 0-AOI-26-1, "Response to Fires"), EOIs or Tech Specs. If candidate considers only these Abnormal / Emergency Procedures, this would be the correct answer.

B INCORRECT: Plausible in that ONLY Unit 1 has equipment that has been damaged by the fire.

C INCORRECT: Plausible in that 4 kV Shutdown Board A supplies loads on Unit 1 and Unit 2.

D **CORRECT**: Per Safe Shutdown Instructions, if SSIs are entered for an Appendix R Fire, ALL 3 Units must be scrammed.

KA Justification:

The KA is met because it tests the candidate's ability to determine need to emergency shutdown Units based plant fire on site.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.031 Rev 13 (Attach if not previously provided)

0-SSI-5 Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _______________ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
700000 Generator Voltage and Electric Grid Disturbances / 6	Tier #	1	------
AK2.07 (10CFR 55.41.5)	Group #	1	------
Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following	K/A #	700000AK2.07	
• Turbine/generator control	Importance Rating	3.6	------

Proposed Question: **# 20**

Unit 3 is operating at 80% Reactor Power **AND** the crew has entered 0-AOI-57-1E, "Grid Instability," due to the 500 kV system voltage being at 513 kV. The crew reaches the following step in the procedure:

- **RAISE** reactive power until voltage returns to 520 kV.

Which ONE of the following identifies how to raise reactive power **AND** the 161 kV Capacitor Bank Status that will restore the system voltage in accordance with 0-AOI-57-1E?

A. Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **IN** service.

B. Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **OUT** of service

C. Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the RAISE position; check the 161 kV Capacitor Banks are **IN** service.

D. Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the RAISE position; check the 161 kV Capacitor Banks are **OUT** of service.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - Depress the EHC load set RAISE pushbutton will have no affect on load or voltage at current power levels. Plausible in that raising load would aid in mitigating the grid low voltage condition. Part 2 is correct as required by 0-AOI-57-1E

B INCORRECT: Part 1 and 2 incorrect – 161 kV Capacitor Banks out of service will not aid in restoring system voltage. Plausible in that it is an action directed under certain conditions for Grid Instability in 0-AOI-57-1E

C **CORRECT:** Part 1 correct – Per 0-AOI-57-1E, RAISE reactive power to system voltage returns to 520KV OR UNTIL Generator Reactive Power reaches +200 MVAR, Per 3-OI-47, To adjust GENERATOR MVAR, 3-EI-57-51, in the positive or lagging direction, PLACE GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, in RAISE UNTIL desired MVAR is indicated. Part 2 correct – Per 0-AOI-57-1E, CHECK 161KV Cap Banks are In Service

D INCORRECT: Part 1 is correct and Part 2 is incorrect.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between low system voltage due to Grid Disturbance and Generator Field Voltage Auto Adjust (90P), 3-HS-57-26.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-AOI-57-1E Rev 7 (Attach if not previously provided)

3-OI-47 Rev 91 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036 V.B.13 (As available)

Question Source: Bank # BFN 0801 #20

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0801

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295002 Loss of Main Condenser Vac / 3	Tier #	1	
AK1.03 (10CFR 55.41.10)	Group #	2	
Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM : • Loss of heat sink	K/A #	295002AK1.03	
	Importance Rating	3.6	

Proposed Question: **# 21**

Unit 3 is operating at 28% Reactor Power, when a lightning strike results in a loss of **ALL** Condenser Circulating Water Pumps. Immediate Actions of 3-AOI-100-1, "Reactor Scram," are complete.

Which ONE of the following identifies the **AUTOMATIC** protective actions that will occur?

A. Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

B. MSIV Closure, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

C. Main Turbine trip, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

D. MSIV Closure, Main Turbine trip, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausibility based on misconception that the Main Turbine trip is bypassed at <30% Reactor Power. The subsequent Reactor Scram due to Turbine Trip is what is bypassed at < 30% Reactor Power.

B INCORRECT**:** Plausibility based on misconception that the Main Turbine trip is bypassed at <30% Reactor Power along with misconception that MSIV closure would result from loss of condenser vacuum. See discussion of MSIV Closure in D explanation.

C **CORRECT:** Main Turbine will trip at condenser vacuum of 21.8" Hg. Both Reactor Feed Pump Turbine Trip and Main Turbine Bypass Valve closure occur at 7" Hg Condenser Vacuum.

D INCORRECT: Plausibility based on misconception that MSIV closure would result from loss of condenser vacuum. The automatic functions associated with degrading condenser vacuum primarily exist to prevent condenser overpressurization. Even after all the automatic functions occur, the condenser is still vulnerable to overpressurization with the MSIVs open. Therefore, it is very logical that an automatic isolation of MSIVs would occur under these conditions and thus removing all sources of Nuclear Steam to the condenser. To make a comparison, there are several examples that can be found on NRC exams that utilize MSIV closure in response to High-High MSL Radiation. One could not really even argue that it is plausible because it was a Group 1 isolation previously since most plants eliminated the function so long ago. However, It is plausible because it is logical that an automatic isolation of MSIVs would occur under these conditions. Additionally, this was a distractor suggested by the chief on our previous NRC exam for a loss of condenser vacuum question. Plausibility also based on if Mode Switch is not taken to Shutdown, the MSIVs could close as a result of this transient due to Reactor Pressure < 850 psig with Mode Switch in Run.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the operational implications (Main Turbine trip / RFPT Trip / MT Bypass Valve closure) of loss of heat sink (all the Condenser Circ Water Pumps tripping) as it applies to Loss of Main Condenser Vacuum.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidate must recognize that 3-AOI-100-1 Immediate Actions require Operator to place the Mode Switch to Shutdown. Then, with Mode Switch in Shutdown, recognize MSIV closure at 850 psig is bypassed.

Technical Reference(s): 3-AOI-47-3, Rev. 11 (Attach if not previously provided)
3-OI-47, Rev. 91
OPL171.010, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010 V.B.12 / 23 (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 X

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295014 Inadvertent Reactivity Addition	Tier #	1	------
G2.4.50 (10CFR55.41.10)	Group #	2	------
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	K/A #	295014G2.4.50	
	Importance Rating	4.2	------

Proposed Question: **# 22**

Unit 1 is performing a startup per 1-GOI-100-1A, "Unit Startup." When the Operator At The Controls (OATC) placed the rod movement control switch to the single notch out position for the next control rod, the rod quickly moved 3 notches beyond its intended position. The following indications are received:

- SRM PERIOD, (1-9-5A, Window 20), in alarm
- SRM period indicates 10 seconds on 1-XI-92-7/44A - D

Which ONE of the following completes the statement below?

The OATC is required to __________.

A. **STOP** Control Rod withdrawal **ONLY**.

B. **INSERT** Control Rods until the Reactor is brought subcritical.

C. **SHUT DOWN** the Reactor until a thorough assessment has been performed.

D. **REINSERT** the last Control Rod withdrawn to obtain a stable period greater than 60 seconds.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that 1-GOI-100-1A directs control rod withdraw stopped for low SRM periods

B **CORRECT:** Per 1-ARP-9-5A and GOI-100-1A, IF withdrawing control rods and a period less than 30 seconds is observed, THEN INSERT rods until subcriticality is observed.

C INCORRECT: Plausible in that this is the correct action if a 5 second period indication is observed.

D INCORRECT: Plausible in that this is the correct action for indication of < 60 but >30 second period.

KA Justification:

The KA is met because to successfully answer this question Operator must be able to verify that the SRM Period alarm as a result of the inadvertent reactivity addition is valid based on period indication. Then, recognize the need to insert control rods until the reactor is subcritical in accordance with the ARP.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-ARP-9-5A Rev 16 (Attach if not previously provided)
1GOI-100-1A Rev 23
OPL171.059 Rev 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.059 V.B.5 (As available)

Question Source: Bank # 1006 Audit # 69
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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295022 Loss of CRD Pumps / 1	Tier #	1	------
AA1.01 (10CFR 55.41.7)	Group #	2	------
Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: • CRD hydraulic system	K/A #	295022AA1.01	
	Importance Rating	3.1	------

Proposed Question: **# 23**

Unit 1 is at 100% Reactor Power when Control Rod Drive (CRD) Pump 1A trips. During the start of CRD Pump 1B, the following occurs:

- Control Rod 30-23 moves from position 16 to position 14
- Control Rod 38-31 is selected and moves from position 16 to position 12 without Operator action

Which ONE of the following identifies the required action(s) in accordance with CRD AOIs?

A. Scram the Reactor.

B. Reduce Reactor Power to 90%

C. Insert Control Rod 30-23 **ONLY** to position 00 using CONTINUOUS IN.

D. Insert Control Rods 30-23 **AND** 38-31 to position 00 using CONTINUOUS IN.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with 1-AOI-85-6, if more than 1 CR drifts, insert a reactor Scram Immediately

B INCORRECT: Plausible in that this is correct AOI actions for a single Control Rod Drifting out and unable to insert the control rod.

C INCORRECT**:** Plausible in that this is the correct AOI action for a single Control Rod Drifting in.

D INCORRECT: Plausible in that this is the correct AOI action for a single Control Rod Drifting in if the Reactor Engineer had provided a verbal or written communication on the first control rod drift, after it had been inserted, and there was a subsequent drift. Or if they believed a control rod moving only one notch is not a rod drift but is mispositioned and must be inserted to 00.

KA Justification:

The KA is met because the question tests the candidate's ability to monitor the CRD hydraulic system as it applies to Loss of the in service CRD Pump. Trip of CRD pump requires start of the standby pump. During start of the standby Pump, the CRD Hydraulic system is susceptible to inadvertent control rod drift if flow is raised rapidly or there is significant seat leakage on the in service CRD flow control valve.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Operator must diagnose multiple control rod drifts based on indication and select appropriate action.

Technical Reference(s): 1-AOI-85-5 Rev. 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 V.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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Wtr Lvl / 5	Tier #	1	------
EK2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: • HPCI: Plant-Specific	K/A #	295029EK2.02	
	Importance Rating	3.4	------

Proposed Question: **# 24**

Unit 1 Suppression Pool Level is (+) 5 inches and CST level is 550'.

Which ONE of the following completes the statements below?

HPCI Suction __**(1)**__ automatically transfer to the Suppression Pool.

RCIC Suction __**(2)**__ automatically transfer to the Suppression Pool.

A. **(1)** will
(2) will

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

C. **(1)** will **NOT**
(2) will

D. **(1)** will **NOT**
(2) will **NOT**

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 correct – See explanation B. Part 2 incorrect – See Explanation C.

B **CORRECT**: Part 1 correct – HPCI Suction automatically swaps to suppression pool on high suppression pool level +5.25" or low CST level Elev <552'6". Part 2 correct - RCIC has no automatic transfer from CST to torus.

C INCORRECT: Part 1 incorrect – Plausible in that this would be true for RCIC. Part 2 incorrect – Plausible in that this would be true for HPCI.

D INCORRECT: Part 1 incorrect – See explanation C. Part 2 correct – See Explanation B.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between High Suppression Pool Water Level and HPCI

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.040 Rev. 23 / 1-OI-73 Rev. 17 (Attach if not previously provided)
OPL171.042 Rev. 20

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 V.B.1 (As available)
OPL171.040 V.B.6

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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295034 Secondary Containment Ventilation High Radiation / 9	Tier #	1	------
EA2.02 (10CFR 55.41.10)	Group #	2	------
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION :	K/A #	295034EA2.02	
• Cause of high radiation levels	Importance Rating	3.7	------

Proposed Question: **# 25**

Unit 1 is at 100% Reactor Power with the following system line ups:

- Reactor Building Closed Cooling Water (RBCCW) Pumps 1A **AND** 1B are in service
- Reactor Water Cleanup (RWCU) Pumps 1A **AND** 1B are in service
- Fuel Pool Cooling and Cleanup (FPCC) Pump 1A is in service

Unit 1 Reactor **Scrams AND** the following alarms / indications are received:

- 480 V Shutdown Board 1A is locked out
- RBCCW SURGE TANK LEVEL HIGH, (1-9-4C, Window 6)
- RBCCW EFFLUENT RADIATION HIGH, (1-9-3A, Window 17)
- RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH, (1-9-3A, Window 4)

Which ONE of the following is a potential cause of the alarms?

Leakage into RBCCW from ____.

A. Reactor Recirc Pump seal coolers

B. Fuel Pool Cooling Heat Exchangers

C. Reactor Water Cleanup Pump Seal Coolers

D. Reactor Water Cleanup Non-Regenerative Heat Exchangers

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** With the isolation of RWCU at (+) 2 inches due to the scram and the loss of FPCC due to the lock out of Shutdown Board 1A, this remains the only choice that is not tripped and/or isolated. RBCCW Pump 1B remains in service supplying Reactor Recirc Pump seal coolers. Therefore, this is a potential source of inleakage into RBCCW and source of the high radiations alarms.

B INCORRECT: Fuel Pool Cooling Pump A power supply is from 480 V Shutdown Board 1A which is locked out. Any FPCC Heat Exchanger leakage would result in leakage into the FPCC system and not into RBCCW. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that FPCC is an RBCCW load and with the absence of the power loss, this could be the source of the Radiation Alarms.

C INCORRECT: With a Scram from 100% power, Reactor Level drops less than (+) 2 inches, RWCU is isolated and the Pumps are tripped. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that RWCU is an RBCCW load and with the absence of the isolation, this could be the source of the Radiation Alarms.

D INCORRECT: With a Scram from 100% power, Reactor Level drops less than (+) 2 inches, RWCU is isolated and the Pumps are tripped. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that RWCU is an RBCCW load and with the absence of the isolation, this could be the source of the Radiation Alarms.

KA Justification:

The KA is met because it tests the candidate's ability to assess the status of RBCCW and its loads to determine the cause of high radiation levels indicated in Secondary Containment and Secondary Containment Ventilation.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.047 Rev. 12 (Attach if not previously provided)

1- ARP-9-3A Rev. 40 / 1- ARP-9-4C Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 V.B.3 (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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295036 Secondary Containment High Sump/Area Water Level / 5	Tier #	1	------
EK3.01 (10CFR 55.41.5)	Group #	2	------
Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL :	K/A #	295036EK3.01	
• Emergency depressurization	Importance Rating	2.6	------

Proposed Question: **# 26**

A HPCI Steam Supply leak has resulted in elevated Secondary Containment temperatures **AND** area water levels. HPCI Steam Supply Isolation valves have failed to isolate **AND CANNOT** be manually closed. Two Secondary Containment Water Levels are above their Maximum Safe Value requiring Emergency Depressurization.

Which ONE of the following completes the statement below?

In accordance with EOI-3, "Secondary Containment Control Bases," **ALL** of the following are reasons for requiring Emergency Depressurization with the **EXCEPTION** of ________.

A. placing the primary system in the lowest possible energy state

B. rejecting decay heat to the suppression pool, rather than secondary containment

C. reducing driving head and flow of primary systems that are unisolated and discharging into secondary containment

D. allowing access into the Reactor Building by the Emergency Response Organization to locate and manually isolate the leak

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

B INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

C INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

D **CORRECT**: This is NOT one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

KA Justification:

The KA is met because the question test knowledge of the reasons for Emergency Depressurization as it applies to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVELS.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Justification:

Technical Reference(s): OPL 171.204 Rev. 7 (Attach if not previously provided)

EOIPM 0-V-E Rev. 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

500000 High Containment Hydrogen Concentration

EK2.09 (10CFR 55.41.7)

Knowledge of the interrelations between HIGH CONTAINMENT HYDROGEN CONCENTRATIONS the following:

- Drywell nitrogen purge system

	RO	SRO
Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	500000EK2.09	
Importance Rating	3.0	------

Proposed Question: **# 27**

Unit 2 was operating at 100% Reactor Power when a LOCA occurred. Plant conditions are as follows:

- Drywell H_2 is 3% increasing
- Drywell O_2 is 4% increasing
- Suppression Chamber H_2 is 2% steady
- Suppression Chamber O_2 is 3% steady

Which ONE of the following completes the statement below?

Based on the above conditions, Nitrogen must be lined up to _______.

A. the Drywell

B. the Suppression Chamber

C. the Drywell **AND** Suppression Chamber

D. **NO** primary containment area; the Primary Containment EOI entry condition for hydrogen concentration has **NOT** been exceeded

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** 2-EOI-2 directs monitoring and controlling Drywell and Suppression Chamber, H2 at or below 2.4% AND O2 at or below 3.3%. The Drywell is above both values. 3% H2 in the Drywell is greater than 2.3%, the minimum detectable value. 2-EOI Appendix 14A states to continue in the procedure when H2 or O2 concentration(s) are increasing. The stem states both are increasing in the Drywell. It then directs the operator to determine which area has the highest H2 or O2 concentrations and directs adding nitrogen to that area to reduce the concentration(s).

B INCORRECT: Suppression Chamber H2 and O2 are below the control limits, NO change is occurring, and lower than the Drywell. Plausible if the candidate doesn't know the control parameter values. H2 value is below the BFN min detectable value of 2.3%.

C INCORRECT: You never add to both areas at once. Procedure adds to one area at a time. Plausible since both areas have elevated H2 and O2 concentrations.

D INCORRECT: Procedure addresses correcting area before 3%. Drywell is above control parameters and increasing. Candidate may not know the EOI entry condition for primary containment hydrogen concentration.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between elevated Primary Containment Hydrogen levels and Nitrogen makeup to Containment.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. The RO has to know the primary containment entry condition for high hydrogen concentration and deduce which area has the worst degrading conditions based on that fact.

Technical Reference(s): 2-EOI-2 Rev 10, OPL171.032 Rev 12 (Attach if not previously provided)
2-EOI-Appendix 14A Rev 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
203000 RHR/LPCI: Injection Mode (Plant Specific)	Tier #	2	------
K3.04 (CFR 41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following:	K/A #	203000K3.04	
• Adequate core cooling	Importance Rating	4.6	------

Proposed Question: **# 28**

An accident occurred on Unit 2 **AND** resulted in the following conditions:

- Reactor water level indicates (–) 200 inches on Post Accident Range
- Reactor pressure is 400 psig
- **ALL** RHR / LPCI are lost
- **ONLY** ONE CRD Pump **AND** ONE Core Spray pump are running

Which ONE of the following completes the statement below?

Adequate core cooling ______.

[REFERENCE PROVIDED]

A. does **NOT** exist

B. is provided by Spray Cooling

C. is provided by Steam Cooling

D. is provided by Core Submergence

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: is incorrect because adequate core cooling exists. The candidate that fails to correct fuel zone level would believe that the core is no longer adequately cooled.

B INCORRECT: is incorrect because reactor pressure is too high for CS to inject. Plausible in that candidate may fail to recognize reactor pressure greater than the shutoff head (330 psig) of the CS pump.

C INCORRECT: is incorrect because the core is submerged with actual level above top of active fuel.

D **CORRECT:** The indicated parameter place corrected water level above TAF. With water level above TAF, adequate core cooling is assured by submergence.

KA Justification:

The KA is met because the question tests knowledge of the affect of Loss of RHR / LPCI on adequate core cooling.

Question Cognitive Level:

Question is rated as C/A because it involves the multi-part mental process of assembling, sorting, and use reference to solve a problem.

Technical Reference(s): OPL171.201 Rev. 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: 2-LI-3-52/62 Correction Curve

Learning Objective: OPL171.201 V.B.10 (As available)

Question Source:

Bank #	CNP 08 #17	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam Cooper 2009

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
205000 Shutdown Cooling	Tier #	2	------
G2.2.22 (10CFR 55.41.5)	Group #	1	------
Knowledge of limiting conditions for operations and safety limits.	K/A #	205000G2.2.22	
	Importance Rating	4.0	------

Proposed Question: **# 29**

Unit 1 is in Mode 4 with RHR Pump 1B in Shutdown Cooling.

Which ONE of the following completes the statements below?

In accordance with Tech Spec 3.5.2, “ECCS - Shutdown,” RHR Pump 1B __**(1)**__ Operable for the ECCS function**.**

The **MAXIMUM** allowed RCS cooldown rate per Tech Spec 3.4.9, “RCS Pressure and Temperature (P/T) Limits,” is __**(2)**__ .

A. **(1)** is
(2) 100°F per hour

B. **(1)** is
(2) 100°F in any one hour period

C. **(1)** is **NOT**
(2) 100°F per hour

D. **(1)** is **NOT**
(2) 100°F in any one hour period

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 correct – See explanation B. Part 2 incorrect – See explanation C.

B **CORRECT:** Part 1 correct - Per Tech Spec 3.5.2, A LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode. Part 2 correct – per Tech Spec 3.4.9, RCS cooldown shall be ≤ 100° F in any one hour period

C INCORRECT: Part 1 incorrect – Plausible in that ECCS systems are normally required to start, align and inject in response to a system initiation signal to be considered operable. The provision to allow manual realignment is an exception for the conditions. Part 2 incorrect – Plausible in that this is very similar to the wording in Tech Spec 3.4.9, and is a common misconception that the cooldown or heat up is “per hour”.

D INCORRECT: Part 1 incorrect – See explanation C. Part 2 correct – See explanation B.

KA Justification:

The KA is met because the question tests knowledge of limiting conditions for operations associated with Shutdown Cooling.

Question Cognitive Level:

Question is rated as C/A because it involves the multi-part mental process of assembling, sorting, and using knowledge and its meaning to solve a problem.

Technical Reference(s): U1 TS 3.4-21 Amm 234 (Attach if not previously provided)

U1 TS 3.4-24 Amm 234/ 3.4-26 Amm 256

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
206000 High Pressure Coolant Injection System	Tier #	2	
A3.05 (CFR: 41.7)	Group #	1	
Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including:	K/A #	206000A3.05	
• Reactor water level:	Importance Rating	4.3*	

Proposed Question: **# 30**

Unit 1 was operating at 100% Reactor Power; when the reactor scrammed on low RPV water level and HPCI auto started, which resulted in the following conditions:

- RPV water level lowered to (-) 50 inches and is currently (+) 55 inches and slowly lowering.

Which ONE of the following, is the **FIRST** condition that would cause an **AUTOMATIC** restart of HPCI?

A. Level lowers to (+) 27 inches.

B. Level lowers to (+) 2 inches.

C. Level lowers to (-) 45 inches.

D. Drywell Pressure greater than 2.45 psig.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With level at (+) 27 inches, the level 8 (+) 51 inches signal will be clear. However, the Level 8 Turbine Trip will still be sealed in, unless manually reset. The candidate may select this if he/she doesn't realize the turbine trip relay seals itself in, and needs to be manually reset. Also (+) 27 is a recognizable value; the set point for the Reactor Level Low alarm.

B INCORRECT: HPCI does **NOT** initiate on a Level 3 signal, (+) 2 inches. HPCI will **NOT** restart, if reactor water level lowers to this value, because of the sealed in Level 8 Turbine Trip. Level 3 is below Level 8 and the candidate may select this as a safe value. PCIS isolations and other events happen at Level 3. HPCI could be restarted with this condition, if the Level 8 reset pushbutton was depressed, on the control room panel.

C **CORRECT:** HPCI will initiate on a Level 2 signal, (-) 45 inches, even though the Level 8 trip, (+) 51 inches, has **NOT** been manually reset. The Level 2 signal opens contacts that de-energize the Level 8 trip relay, which enables the HPCI Turbine to auto restart.

D INCORRECT: A drywell pressure of 2.45 psig is a normal HPCI initiation signal, and the signal seals in. However, the HPCI Turbine Trip is sealed in and will **NOT** reset on this initiation signal. Since this is an initiation signal, the candidate may think the HPCI Turbine will automatically restart.

KA Justification:

K/A is matched because question is on the HPCI system and monitoring automatic operation, based on water level conditions. The question asks what water level condition will allow HPCI to auto restart, based on the conditions of the stem.

Question Cognitive Level:

The candidate must know several facts: HPCI initiates on Level 2 (-) 45 inches reactor water level and on High Drywell Pressure (+) 2.45 psig. The stem also states level is (+) 55 inches and the candidate must determine that water level is above level 8 (+) 51 inches. The candidate must also know that the HPCI level 8 Turbine Trip Logic seals in and does not automatically reset. Operator action is required to manually reset it, unless Level 2 is reached. The sealed in Level 8 HPCI Turbine Trip will NOT allow the sealed in HPCI Initiation Signal Hi Drywell press to restart the system unless the Trip is manually reset or level again lowers to Level 2. Level 2 contacts will open and de-energize the L8 Turbine Trip relay, which will facilitate an automatic restart. To solve the problem posed by the question, the candidate must use a multi-part mental process to assemble, sort, and integrate parts of the HPCI and HPCI Logic systems.

Technical Reference(s): OPL171.042 Rev 20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3.c (As available)

Question Source:	Bank #	Fermi 2	
	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 **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments: References attached.

Examination Outline Cross-reference:	Level	RO	SRO
209001 Low Pressure Core Spray System	Tier #	2	------
K1.07 (10 CFR 55.41.2 to 41.9)	Group #	1	------
Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following:	K/A #	209001K1.07	
• D.C. electrical power	Importance Rating	2.5	------

Proposed Question: **# 31**

Unit 2 was operating at 100% Reactor Power, when a plant event resulted in a reactor scram **AND** loss of 250 VDC RMOV BD 2A. Degrading plant conditions have resulted in the following:

- Reactor Pressure is 325 psig and stable
- A few minutes later, Drywell Pressure is 2.8 psig

Based on the above conditions, which ONE of the following predicts how Core Spray will be affected by the bus loss?

A. **ALL** Core Spray pumps will start **AND ALL** injection valves will open.

B. **ONLY** the Loop 1 Core Spray pumps will start **AND** Loop 1 injection valves will open.

C. **ONLY** the Loop 2 Core Spray pumps will start **AND** Loop 2 injection valves will open.

D. **NO** Core Spray pumps will start **AND NO** injection valves will open.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Loop 2 pumps will not start and injection valves will not open. Candidate misconception that logic failure causes valves to fail open and pump start will NOT be affected.

B **CORRECT**: Loop 1 pumps will start and injection valves will open. SYS I Initiation Logic is still energized.

C INCORRECT: Loop 2 pumps will not start and injection valves will not open. Candidate misconception that logic failure causes valves to fail open and pump start will NOT be affected. Candidate misconception that 250 VDC RMOV BD 2A is a division 1 feed and affects Loop 1 pumps and valves.

D INCORRECT: Loop 1 pumps will start and injection valves will open. Candidate misconception that there is only one logic system for both loops of Core Spray so both would be affected and Loop 2 logic would be for UNIT 2.

KA Justification:

K/A requires cause effect relationship between Core Spray System and DC power. Question is about Core Spray system and the loss of DC power to one portion of its initiation logic and its effect on the system.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must know the power supply to the Core Spray loop 2 logic and the effects of its loss. He/she must understand the system and logic interrelationships.

Technical Reference(s): OPL171.045 Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OPL171.045 Obj 4.d (As available)

Question Source: Bank #

Modified Bank # VY 2007 NRC Q6 (Note changes or attach parent)

New

Question History: Last NRC Exam Vermont Yankee 2007

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
211000 Standby Liquid Control System	Tier #	2	------
A2.07 (10CFR 55.41.5)	Group #	1	------
Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	211000A2.07	
• Valve closures	Importance Rating	2.9	------

Proposed Question: **# 32**

Unit 1 is executing 1-EOI-1, "RPV Control," due to a Scram **AND** an ATWS. The Unit Operator (UO) is directed to inject Standby Liquid Control (SLC) per 1-EOI Appendix 3A, "SLC Injection."

The UO places the SLC Pump control switch in the 'START-A' position.

Given the following plant conditions:

• SLC SQUIB VALVE CONTINUITY LOST, (1-9-5B, Window 20)	Extinguished
• SQUIB VALVE A and B CONTINUITY, blue lights on Panel 1-9-5	Illuminated
• SLC Pump 1A red light	Illuminated

Which ONE of the following describes the status of SLC **AND** the correct action(s) to take as stated in 1-EOI Appendix 3A?

A. **ONE** squib valve has fired.
Start SLC Pump 1B, **AND** verify proper operation.

B. **NEITHER** squib valve has fired.
Start SLC Pump 1B, **AND** verify proper operation.

C. **ONE** squib valve has fired.
Verify proper system operation by observing the SLC tank level lowering by ~1% per minute.

D. **BOTH** squib valves have fired.
Verify proper system operation by observing the SLC tank level lowering by ~1% per minute.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that 'A' pump did start by indication of RED light illuminated. A candidate may believe this is an indication that one squib valve has fired; however, neither squib valve has fired; as indicated by the lack of the alarm and the blue lights being lit. The squib valves are arranged in Parallel, so 1 firing would allow injection into RPV. Starting 'B' would allow the squib valves to be fired from the other primer.

B **CORRECT:** 'A' pump did start by indication of RED light illuminated. Neither squib valve has fired; as indicated by the lack of the alarm and the blue lights being lit. Starting 'B' would allow the squib valves to be fired from the other primer.

C INCORRECT: Plausible in that 'A' pump did start by indication of RED light illuminated. A candidate may believe this is an indication that one squib valve has fired. Verifying proper system operation by observing SLC tank level lowering would be a step in the EOI appendix to be performed.

D INCORRECT: Plausible in that a candidate may mistake squib valve blue lights being illuminated to mean that the valves have fired. Which is how the TIP squib valve indication works. Verifying proper system operation by observing SLC tank level lowering would be a step in the EOI appendix to be performed.

KA Justification:

The KA is met because the question tests the ability to predict the impact of valve closures on the SLC System. Based on the indications provided, candidate must conclude that following system initiation both Squib Valves remain closed and recognize the impact on SLC Injection. Based on the Squib Valves failing to open, the candidate must use 1-EOI-1 Appendix 3A to correct the consequences of this abnormal condition.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidate must diagnose the system condition based on indications provided and then determine appropriate action to take to correct the abnormal condition.

Technical Reference(s): 1-EOI Appendix 3A rev 0 (Attach if not previously provided)

OPL171.039 rev 16 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.4 / V.B.5 (As available)

Question Source: Bank # BFN 0801 #33

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0801

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: The 'A' SLC pump has started and neither squib valve has fired as indicated by the lack of the alarm and the blue lights are still lit. The proper action iaw EOI-app 3A is to start the other pump and verify proper operation.

Examination Outline Cross-reference:	Level	RO	SRO
212000 Reactor Protection System	Tier #	2	-----
A4.03 (10 CFR 55.41.7)	Group #	1	-----
Ability to manually operate and/or monitor in the control room:	K/A #	212000A4.03	
• Provide manual select rod insertion	Importance Rating	3.9	-----

Proposed Question: **# 33**

Unit 2 was operating at 100% Reactor Power, when the plant experienced a complete loss of the Control Air system. The following plant conditions exist:

- **ALL** eight Scram Solenoid Group A/B Logic Reset Lights are **NOT** lit
- Recirc Pumps are Tripped
- Reactor Power is 20%

You are the OATC and have been directed to perform 2-EOI Appendix 1D, "Insert Control Rods Using Reactor Manual Control System" (RMCS).

Which ONE of the following completes the statement below?

Verify CRD Pump operating, ____(1)____ , direct manually opening CRD Flow Control Valve (2-FCV-85-11A or B), verify Mode Switch in SHUTDOWN, bypass the Rod Worth Minimizer, CRD Power Switch ON, select control rod, **AND** place CRD ____(2)____.

A. **(1)** reset ARI
(2) Control Switch in ROD IN, until green 00 is lit, on the four rod display

B. **(1)** reset ARI
(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

C. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586
(2) Control Switch in ROD IN, until the green 00 is lit, on the four rod display

D. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586
(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: A loss of Control Air occurred, so scram and ARI cannot be reset. Also CRD Notch Override Switch is placed in Emergency In, in an ATWS. Procedure directs insert until movement stops. Candidate misconception that scram and ARI can be reset with NO Control Air available. Also misconception that CRD Control Switch is used in an ATWS when driving rods. NOT used due to RMCS settle function requirements between rods, would delay rod insertion in this emergency.

B INCORRECT: A loss of Control Air occurred, so ARI and scram cannot be reset. Part 2 is correct; the procedure directs insert until movement stops and use of Notch Override Switch in EMERGENCY IN until rod stops moving. Candidate misconception that scram and ARI can be reset with NO Control Air available.

C INCORRECT: A loss of Control Air occurred. Part 1 is correct because cannot reset scram or ARI. Candidate misconception that CRD Control Switch is used in an ATWS when driving rods. NOT used due to RMCS settle function requirements between rods, would delay rod insertion in this emergency. CRD Notch Override Switch is placed in Emergency In to insert the control rod, in an ATWS

D **CORRECT:** A loss of Control Air occurred. Scram and ARI cannot be reset because no air pressure. Charging water shutoff valve needs to be closed to direct water from Charging header to Drive Water Header to move rods. The CRD Flow Control Valve has lost air and needs to be manually opened to provide Drive Water Pressure to drive control rods. Emergency In is used to bypass the settle function on the Reactor Manual Control Sys, so the control rods can be inserted without waiting between rod selections, therefore taking less time to insert in the ATWS emergency.

KA Justification:

The K/A is matched because the question and K/A require how to manually select, insert, and determine (monitor) when the control rods are inserted.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. The candidate must deduce that an ATWS has occurred. He/she must determine that the loss of Control Air caused the scram and Recirc Pump Trip. The loss of Control Air will not allow reset of the scram or ARI. It complicates control rod movement because of loss of air to the CRD Flow Control Valve. Because of the ATWS, control rod movement will be with the ROD Notch Override Switch instead of the CRD Control Switch.

Technical Reference(s): 2-EOI Appendix 1D Rev 6 (Attach if not previously provided)

2-AOI-32-2 Rev 32

Proposed references to be provided to applicants during examination: None

Learning Objective: V.B.9 (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	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215003 Intermediate Range Monitor (IRM) System	Tier #	2	------
K2.01 (10 CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following:	K/A #	215003K2.01	
• IRM channels/detectors	Importance Rating	2.5	------

Proposed Question: **# 34**

Unit 2 is performing a startup with the following conditions:

- Mode Switch is in STARTUP
- Reactor is critical
- IRMs are steady on Range 2

Which ONE of the following identifies the IRM power source **AND** the effect of a loss of power to a single IRM?

	IRM Power Source	**Effect of Power Loss to IRM**
A.	24 VDC Battery	Rod Block **ONLY**
B.	24 VDC Battery	Rod Block **AND** Half Scram
C.	250 VDC Battery	Rod Block **ONLY**
D.	250 VDC Battery	Rod Block **AND** Half Scram

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: An INOP half scram is also processed, as well as a rod block. Candidate misconception that scram function bypassed on range 2.

B **CORRECT:** 24 VDC supplies IRM detector voltage. With a loss of power, the detector will indicate downscale and receive an INOP trip. The INOP trip enforces both a rod block and a half scram on the corresponding RPS channel.

C INCORRECT: 24 VDC supplies IRM detector voltage. An INOP half scram is also processed. Candidate misconception that 250 VDC supplies IRMs. It does supply the neutron monitoring battery chargers.

D INCORRECT: 24 VDC supplies IRM detector voltage. Candidate misconception that 250 VDC supplies IRMs. It does supply the neutron monitoring battery chargers.

KA Justification:

K/A is matched because the question asks for power supply to the IRMs and affect of loss of the power supply. K/A asks for knowledge of electrical power supply to the IRMs channels/detectors.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.020 Rev 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.11 (As available)

Question Source: Bank #

Modified Bank # Nine Mile 2 /Q23 (Note changes or attach parent)

New

Question History: Last NRC Exam Nine Mile 2 / 2008

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor (SRM) System	Tier #	2	
K5.01 (10 CFR 55.41.5)	Group #	1	
Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : • Detector operation	K/A #	215004K5.01	
	Importance Rating	2.6	

Proposed Question: **# 35**

Which ONE of the following completes the statement below?

The applied voltage to the SRM detector is __**(1)**__ than the applied voltage used for the IRM detector **AND** the SRM electrode generates an electrical signal __**(2)**__ proportional to neutron flux in the core.

A. **(1)** lower
(2) directly

B. **(1)** higher
(2) directly

C. **(1)** lower
(2) inversely

D. **(1)** higher
(2) inversely

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: SRM voltage is higher. Candidate misconception that SRM detectors detect lower power therefore the voltage detector power requirement is lower.

B **CORRECT:** The SRM (IRM) detector is a fission chamber that has an applied voltage to the electrode of approximately 350 (100) volts. The operating chamber is pressurize with Argon to about 213 (17) psia. They generate an electrical signal proportional to the neutron flux level in the core.

C INCORRECT: SRM voltage is higher and the signal is not inversely proportional. Candidate misconception that SRM detectors detect lower power therefore the voltage detector power requirement is lower. Candidate misconception that Campbeling correction (square root effect) is used by the SRM, and this makes the signal inversely proportional.

D INCORRECT: the signal is not inversely proportional. Candidate misconception that Campbeling correction (square root effect) is used by the SRM, and this makes the signal inversely proportional.

KA Justification:

K/A is met by question asking knowledge of the SRM detector operation. RO knowledge Task. Memory knowledge because RO must recall facts about SRM detector operation.

Question Cognitive Level:

The question tests for the total recall of discrete facts or bits of information, for a single system.

Technical Reference(s): OPL171.019 Rev 13 (Attach if not previously provided)
OPL171.020 Rev 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.D.2 (As available)

Question Source:

Bank #	Brunswick 07 #12	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam Brunswick 2007

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor (SRM) System	Tier #	2	------
K5.03 (10 CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM :	K/A #	215004K5.03	
• Changing detector position	Importance Rating	2.8	------

Proposed Question: **# 36**

A plant start up on Unit 3 is in progress. A control rod block has occurred. The following nuclear instrument indications are noted:

	SRM A	SRM B	SRM C	SRM D
Position	Full in	Mid-position	Mid-position	Full in
Counts (CPS)	$9.5x10^3$	125	150	$8.0x10^3$

IRM A	IRM B	IRM C	IRM D	IRM E	IRM F	IRM G	IRM H
25/125	15/125	35/125	55/125	75/125	75/125	30/125	25/125
Range 3	Range 2	Range 3	Range 3	Range 2	Range 2	Range 3	Range 3

Which ONE of the following identifies the **MINIMUM** action needed to clear the ROD WITHDRAWAL BLOCK?

A. Insert SRM B **ONLY**

B. Insert SRM B **AND** SRM C

C. Range up on IRM B **AND** IRM F to range 3

D. Range up on IRM E **AND** IRM F to range 3

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: SRM RETRACT NOT PERMITTED will alarm and cause a rod block with SRM counts <145cps with associated IRMs ≤ Range 2 and the Detector not Full In.

B INCORRECT: Plausible in that with SRM C Not Full in and associated IRM E not on range 3, candidate may believe that it must also be inserted to clear the Rod Block. However, although SRM C is not full in, it is above the Rod Block set point of 145 cps so the Rod Block is bypassed.

C INCORRECT: Plausible in that it would clear the Control Rod Block from SRM B. However, it would result in IRM B causing a rod block due to IRM downscale.

D INCORRECT: Plausible in that ranging up IRM E and F would not result in an IRM downscale rod block. However, a rod block would remain with IRM B still on range 2.

KA Justification:

K/A is matched because in the question operational conditions/implications have arisen from the mis-positioning of the SRM detectors. The candidate must determine which detector is causing the conditions and based on his/her knowledge resolve the situation. Knowledge involves recognizing the interaction between the SRM/IRM systems, including consequences and implications.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-OI-92 Rev. 14 (Attach if not previously provided)
OPL171.019 Rev 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.019 V.B.8 (As available)

Question Source: Bank #
Modified Bank # BFN 1006 #37 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215005 APRM / LPRM	Tier #	2	
A3.08 (10CFR 55.41.7)	Group #	1	
Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including:	K/A #	215005A3.08	
• Control rod block status	Importance Rating	3.7	

Proposed Question: **# 37**

Unit 2 APRM Channel 3 has a total of 18 LPRM inputs.

Which ONE of the following statements identifies the expected response to this condition?

A. The APRM will produce a Rod Block signal **ONLY**.

B. **NO** Rod Block **OR** Reactor Scram signals are generated.

C. The APRM will produce a Rod Block signal **AND** a Scram signal input to **EACH** 2/4 logic voter module.

D. The APRM will produce a Rod Block signal **AND** a Scram signal input to **ITS RESPECTIVE** 2/4 logic voter module **ONLY**.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: If the number of un-bypassed LPRM inputs exceeds the minimum number required in the APRM average (<20 total or <3 per level), an APRM INOP condition is applied. This results in a Rod Block only - manual trip must be inserted for inoperable condition.

B INCORRECT: Plausibility based on misconception that since no Reactor Scram signal is generated with this Inop condition, likewise, no Control Rod Block is generated. Also plausible that the candidate may believe the minimum number of LPRM inputs is still available and conditions are not met for Rod Block or Scram Signal.

C INCORRECT: Plausible in that < 20 LPRM inputs to an APRM results in INOP Condition. ALL other APRM Inop signals do result in an APRM Trip. This would be the correct answer for any other APRM Inop Signal.

D INCORRECT**:** Plausibility based on the misconception that a Scram Signal would result with < 20 LPRMs input into the APRM and that the resultant scram signal would input only into associated logic voter module.

KA Justification:

The KA is met because the question tests ability to monitor automatic operations of the Average Power Range Monitoring System including Control rod block status and scram signal input to voter logic given less than the required 20 LPRM inputs into an APRM.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.148 Rev 12 (Attach if not previously provided)
2-OI-92B Rev. 38

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.148 V.B.7/31 (As available)

Question Source:

Bank #	OPL171.148 #58
Modified Bank #	
New	
Last NRC Exam	

(Note changes or attach parent)

Question History:

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
217000 Reactor Core Isolation Cooling System (RCIC)	Tier #	2	------
K2.02 (10 CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following:	K/A #	217000K2.02	
• RCIC initiation signals (logic)	Importance Rating	2.8	------

Proposed Question: **# 38**

Unit 2 experienced a loss of 250 VDC RMOV BD 2B

Which ONE of the following statements describes the operation of the RCIC system?

A. RCIC will NOT automatically isolate.

B. RCIC will NOT automatically initiate.

C. The RCIC Flow Controller fails downscale.

D. ONLY the manual isolation is functional.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: ONLY Channel/Bus A isolation Logic is inop. Channel/Bus B Isolation is still functional, an auto isolation can occur. Candidate could confuse isolation logic power supplies.

B **CORRECT:** 250 VDC RMOV BD 2B supplies the Auto Initiation Logic and Auto Channel/Bus A Isolation Logic.

C INCORRECT: The RCIC Flow Controller is fed from the Div 1 ECCS ATU Inverter. Loss of 250 VDC will NOT affect the flow controllers operation. HPCI and RCIC system components and power supplies are easily confused by the examinees.

D INCORRECT: The RCIC Channel/Bus A Isolation Logic is inop. The Manual isolation is ONLY in the Channel/Bus A Isolation Logic. Manual Isolation is ONLY functional if an auto initiation of RCIC occurs, and the Auto Initiation Logic is inop., so this will not be functional. HPCI and RCIC system components and power supplies are easily confused by the examinees.

KA Justification:

The KA is met because the question tests the knowledge of electrical power supply to RCIC initiation logic. The RMOV Board lost in the stem is the power supply to the RCIC Initiation Logic.

Question Cognitive Level:

This question is rated as Fundamental Knowledge because it requires recall of discrete information and is a memory or low cognitive question.

Technical Reference(s): OPL171.040 Rev 23 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040 Obj V.B.7 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
217000 Reactor Core Isolation Cooling System (RCIC)	Tier #	2	
K2.04 (10CFR 55.41.7)	Group #	1	
Knowledge of electrical power supplies to the following: • Gland seal compressor (vacuum pump)	K/A #	217000K2.04	
	Importance Rating	2.6	

Proposed Question: **# 39**

Which ONE of the following completes the statement below?

The power supply to the Unit 2 RCIC Vacuum Pump is ________.

A. 250 VDC RMOV BD 2A

B. 250 VDC RMOV BD 2C

C. 480 VAC RMOV BD 2A

D. 480 VAC RMOV BD 2B

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

B **CORRECT:** 250 VDC RMOV BD 2C is the power supply to the RCIC Vacuum Pump. See Attached Electrical Lineup Checklist.

C INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

D INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

KA Justification:

The KA is met because the question tests candidate knowledge of power supplies to RCIC Vacuum Pump. Level of difficulty is compounded by the similarities of HPCI and RCIC in conjunction with the complex electrical distribution system at BFN. HPCI is a Div II System with 'B' Logic as the primary logic; but it comes from an 'A' Board. RCIC is the opposite – 'A' Logic from a 'B' Board. This often creates confusion between the power supplies for the two systems.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 2-OI-71, Rev. 61 / 2-OI-71 Att. 3 Rev. 58 (Attach if not previously provided)

OPL171.040 Rev. 23

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
218000 ADS	Tier #	2	------
G2.1.7 (10CFR 55.41.5)	Group #	1	------
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	218000G2.1.7	
	Importance Rating	4.4	------

Proposed Question: **# 40**

Unit 2 was operating at 100% Reactor Power with RHR Pump 2D tagged out of service. A Loss of Coolant Accident with a subsequent Loss of Off Site Power has resulted in the following plant conditions:

- Reactor Water Level is (-)125 inches
- Drywell Pressure is 4.1 psig
- A **AND** C 4KV Shutdown Boards are de-energized

Which ONE of the following identifies the **MINIMUM** action(s), if any, that will prevent the Automatic Depressurization System (ADS) from an Auto-Initiation?

A. **NO** action is required

B. Place **ONLY** ADS Logic Inhibit Switch 'A' to INHIBIT

C. Place **ONLY** ADS Logic Inhibit Switch 'B' to INHIBIT

D. Place **BOTH** ADS Logic Inhibit Switches to INHIBIT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: RHR Pump C running meets Pump running permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. RHR C Pump is running and NO Core Spray Pumps are running.

B INCORRECT: Plausible in that different combinations of ECCS Pumps operating meet the pump running permissive for different ADS logic channels.

C INCORRECT: Plausible in that different combinations of ECCS Pumps operating meet the pump running permissive for different ADS logic channels.

D **CORRECT**: RHR Pump C running meets Pump running permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. RHR C Pump is running and NO Core Spray Pumps are running

KA Justification:

The KA is met because the question tests candidates' ability to evaluate plant performance and make operational judgments for the ADS System based on operating characteristics, reactor behavior, and instrument interpretation including Reactor Level, Drywell Pressure and Electrical Distribution indications. Based on those indications, candidate must make operational judgment regarding the status of ADS logic.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.043 Rev 13 (Attach if not previously provided)

2-OI-1 Rev. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.043 V.B.4 (As available)

Question Source: Bank #

Modified Bank # BFN 1006 #40 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off	Tier #	2	
K4.02 (10 CFR 55.41.7)	Group #	1	
Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following:	K/A #	223002K4.02	
• Testability	Importance Rating	2.7	

Proposed Question: **# 41**

Which ONE of the following completes the statement below?

A trip of **BOTH** division 1 (A, C) Reactor Water Cleanup Suction Isolation Valves low level sensor relay(s) within a logic trip channel will cause a __**(1)**__ isolation **AND** __**(2)**__ closure.

A. **(1)** half
(2) NO valve

B. **(1)** half
(2) inboard valve

C. **(1)** full
(2) inboard valve

D. **(1)** full
(2) inboard **AND** outboard valve

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Typical PCIS logic is designed so each valve has 2 trip channels, each containing 4 level sensor relays two from division 1 (A and C contacts in series) and two from division 2 (B and D contacts in series) with both sets of contacts in parallel. The trip of one or both division 1 low level sensor relays in a single channel will cause a half isolation on the Inbd and Obrd valves and no valve closure. The isolation is said to be half-cocked. A trip of one or both low level sensor relays in each division will cause a full isolation and valve closure. (Inbd and Obrd valves)

B INCORRECT: Half is correct, but no valve closure will occur. It would take a trip of a sensor relay in the other low level sensor division to affect closure.

C INCORRECT: Full is incorrect. There would be no valve closure for the conditions given. Misconception by candidate that a trip of any two sensor relays would cause valve closure.

D INCORRECT: Full is incorrect. Neither valve would move under the given conditions. Misconception of logic operation.

KA Justification:

The K/A is matched because the stem asks for knowledge of how a trip channel is tested and how an isolation does not occur. This is RO level knowledge because Instrument technicians test isolation instrumentation daily in the plant without isolations occurring. Knowledge is covered in lesson plan.

Question Cognitive Level:

Examinee must know discrete bits of information about the system.

Technical Reference(s): OPL171.017 Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
223002 PCIS/Nuclear Steam Supply Shutoff **K4.05 (10CFR 55.41.7)**	Tier #	2	------
Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: • Single failures will not impair the function ability of the system	Group #	1	------
	K/A #	223002K4.05	
	Importance Rating	2.9	------

Proposed Question: **# 42**

Unit 2 is starting up following a refueling outage with Reactor Pressure at 80 psig.

RPS MG Set A has tripped. RPS Distribution Panel A has **NOT** yet been transferred to its alternate source.

The Low-Low-Low Reactor Water Level instrument providing input to PCIS Channel B2 fails downscale.

Which ONE of the following describes the response of MSIVs **AND** Main Steam Line Drains?

A. **ONLY** the Inboard Steam Line Drain valve **AND ALL** MSIVs close.

B. **ONLY** the Outboard Steam Line Drain valve **AND ALL** MSIVs close.

C. Inboard **AND** Outboard Steam Line Drain valves **AND ALL** MSIVs close.

D. Inboard **AND** Outboard Steam Line Drain valves close, **AND ALL** MSIVs remain open.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that Loss of RPS A will close MSL Inboard Drain Valve AND deenergize MSIV AC solenoids. However with B2 failed downscale and RPS A deenergized, both A and B logic are made up to deenergize both AC and DC solenoids and provides an isolation signal to the outboard MSL drain. If B1 channel had failed, this would be the correct answer.

B INCORRECT: Plausibility based on misconception that only outboard will isolate as result of combination of logic power and failure of B2. The inboard valve will close as a result of loss of relay power with loss of RPS A. If RPS B had failed, this would be the correct answer.

C **CORRECT**: Channel B2 tripped would give a Group 1 logic *B/D* tripped, loss of RPS A would remove power from Group 1 logic A/C and result in a full MSIV isolation. A2 (Loss of RPS) and B2 closes outboard steam line drain. Loss of A logic power from RPS A will close the Inboard steam line drains.

D INCORRECT: Plausibility based on misconception that DC Pilot Solenoids would remain energized and therefore MSIVs remain open since either solenoid energized maintains the valves open. If B logic was also powered from 250 VDC, like the DC solenoids, this would be the correct answer.

KA Justification:

The KA is met because the question tests candidate's knowledge of Primary Containment Isolation System design features and interlocks which provide for single failures not impairing the function ability of the system.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-1, Rev. 47 (Attach if not previously provided)

OPL171.017, Rev.15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.017 V.B.3 (As available)

Question Source: Bank # Brunswick 07 #17

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Brunswick 2007

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
239002 SRVs	Tier #	2	------
K1.01 (10CFR 55.41.3)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following:	K/A #	239002K1.01	
• Nuclear boiler	Importance Rating	3.8	------

Proposed Question: **# 43**

During a transient on Unit 1, Reactor Pressure reached 1150 psig.

Which ONE of the following identifies how many SRVs opened?

A. Four

B. Eight

C. Nine

D. Thirteen

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if Reactor Pressure was between 1135 and 1145 psig.

B **CORRECT**: The first two groups open with Reactor Pressure > 1145 psig. Each of these groups has 4 valves.

C INCORRECT: Plausible in that this would be the correct answer if group 2 had 5 SRVs instead of group 3

D INCORRECT: Plausible in that this would be the correct answer if Reactor Pressure was > 1155 psig.

KA Justification:

The KA is met because the question tests the candidates' knowledge of the cause-effect relationship between the Nuclear Boiler and SRVs.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.009, Rev. 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 V.B.2 (As available)

Question Source: Bank # OPL171.009 #3

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 **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control System	Tier #	2	-----
A1.01 (10 CFR 55.41.5)	Group #	1	-----
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including:	K/A #	259002A1.05	
• Reactor water level	Importance Rating	3.8	-----

Proposed Question: **# 44**

Unit 2 Feedwater Level Control System (FWLCS) is operating in 3-Element Control with Narrow Range Level Instruments indicating as follows:

- 2-LT-3-53, LEVEL A, (+) 46 inches
- 2-LT-3-60, LEVEL B, (+) 32 inches
- 2-LT-3-206, LEVEL C, (+) 34 inches
- 2-LT-3-253, LEVEL D, (+) 33 inches

Which ONE of the following completes the statement below?

If 2-LT-3-60, LEVEL B, is manually bypassed, the FWLCS will control Reactor Water Level based on ______.

A. **ONLY** the 2-LT-3-206 instrument

B. **LOWEST** of 2-LT-3-206 **OR** 2-LT-3-253 instruments

C. **AVERAGE** of 2-LT-3-206 **AND** 2-LT-3-253 instruments

D. **AVERAGE** of 2-LT-3-53, 2-LT-3-206, **AND** 2-LT-3-253 instruments

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that if FWLCS selected the middle of the 3 remaining channels when one channel is bypassed, this would be the correct answer.

B INCORRECT: Plausible in that if FWLCS selected the lower of the channels not manually or automatically bypassed, this would be the correct answer.

C **CORRECT:** The average level value is used for the three element control logic. The algorithm validates each level signal by comparing them to the average. Level signals that deviate from the average by more than 8 inches are declared invalid, and are discarded from the average. LT-3-53 deviation is > 8" and is bypassed and LT-3-60 is manually bypassed. If two level signals are BAD or invalid, the algorithm will average the remaining two levels and will control on that value. In this instance the two remaining signals

D INCORRECT: Plausible in that if candidate fails to recognize that 2-LT-3-53, LEVEL A is bypassed due to deviation >8 inches from average, this would be the correct answer.

KA Justification:

The KA is met because the question test candidates' ability to predict and monitor changes in Reactor water level associated with operating the Reactor Water Level Control System. Candidate must recognize that one level channel meets the criteria to be automatically bypassed. Then, when another channel is manually bypassed, candidate must predict how the level control logic will function to monitor for expected changes in Reactor Level.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.012 Rev 14 (Attach if not previously provided)

2-OI-3 Rev 136

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.012 V.B.5 (As available)

Question Source: Bank # BFN 1006 Audit #44

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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: Although this question has been modified from its original bank form to meet the KA, it does not meet the criteria for a significantly modified question and is therefore designated as a Bank Question.

Examination Outline Cross-reference:	Level	RO	SRO
261000 SGTS	Tier #	2	------
K4.05 (10CFR 55.41.7)	Group #	1	------
Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: • Fission product gas removal	K/A #	261000K4.05	
	Importance Rating	2.6	------

Proposed Question: **# 45**

Which ONE of the following completes the statement below?

Standby Gas Treatment System **__(1)__** are designed to remove elemental iodine AND the **__(2)__** are designed to reduce relative humidity to less than 70%.

A. **(1)** HEPA Filters
(2) Moisture Separators

B. **(1)** Carbon Beds
(2) Moisture Separators

C. **(1)** HEPA Filters
(2) Electric Heaters

D. **(1)** Carbon Beds
(2) Electric Heaters

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that HEPA filters function to remove fine particulate matter. Part 2 incorrect – Plausible in that Moisture Separators remove water vapor.

B INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 1 correct – See Explanation D.

D **CORRECT:** Parts 1 and 2 correct - Carbon Beds are designed to remove at least 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190°F. Electric heaters reduce the humidity of the air stream.

KA Justification:

The KA is met because the question tests candidates' knowledge of Standby Gas Treatment System Carbon Bed design criteria which provide for fission product gas removal.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.018 Rev. 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.018 V.B.6 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
262001 A.C. Electrical Distribution	Tier #	2	------
K3.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: • Uninterruptible power supply	K/A #	262001K3.04	
	Importance Rating	3.1	------

Proposed Question: **# 46**

The Unit 1 Unit Preferred Inverter is operating in a normal lineup, when a loss of off-site power **AND** a failure of DG "A" to start occurs.

Which ONE of the following completes the statement below?

The Unit Preferred Inverter is powered from ____________________ .

A. 480V RMOV BD 1A

B. 250 VDC Battery Board 4

C. 250 VDC Battery Board 5

D. the Unit Preferred Transformer

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The UPS Rectifier/Inverter is normally powered from the 480V RMOV BD 1A, but it is NOT energized based on the conditions given. Plausible because the candidate may believe that 480V RMOV BD supplied by auto transfer to DG "B".

B INCORRECT: Battery Board 4 is the alternate DC supply to the inverter and would have to be manually shifted to supply it. Plausible because easily confused with Battery Board 5 and it is the normal supply to one of the MMG's. MMG's are also a Unit Preferred System.

C **CORRECT:** Loss of off-site power and a failure of DG "A" to start would result in no power to 4kV SD BD 1A, 480V SD BD 1A, and 480V RMOV BD 1A, which is the Normal supply to the Unit Preferred Rectifier/Inverter. The UPS would automatically shift to 250 VDC Battery Board 5 supplying the inverter, when the diode in the inverter is no longer reversed biased by the rectifier output.

D INCORRECT: The Unit Preferred Transformer is supplied by 480V RMOV BD 1A, which is also the normal supply to the Rectifier/Inverter. This RMOV Board has no power based on the given conditions. IF it were powered, it would have to be manually shifted to supply the static inverter. Plausible because candidate may believe it is powered from 480V RMOV Bd "B".

KA Justification:

The KA is met because the question tests knowledge of the effects of loss of offsite power and failure of EDG A has on the Unit 1 Unit Preferred Inverter which is an uninterruptible power supply.

Question Cognitive Level:

This question is low cognitive or memory question.

Technical Reference(s): OPL171.102 Rev 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.2.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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
262002 UPS (AC/DC)	Tier #	2	-----
A2.02 (10CFR 55.41.5)	Group #	1	-----
Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • Over voltage	K/A #	262002A2.02	-----
	Importance Rating	2.5	

Proposed Question: **# 47**

Which ONE of the following completes the statements below?

The 1001 **AND** 1003 breaker from Unit 2 Unit Preferred System (UPS) Motor-Motor-Generator (MMG) set will trip on __**(1)**__ at the output of the MMG.

In accordance with 2-AOI-57-4, "Loss of Unit Preferred," if UPS is lost, the crew must __**(2)**__.

A. **(1)** under frequency **ONLY**
(2) take manual control of Master Feedwater Level Controller

B. **(1)** under frequency **ONLY**
(2) verify Reactor Feedwater Control System is maintaining Reactor Water Level

C. **(1)** under frequency **OR** overvoltage
(2) take manual control of Master Feedwater Level Controller

D. **(1)** under frequency **OR** overvoltage
(2) verify Reactor Feedwater Control System is maintaining Reactor Water Level

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that under frequency ONLY at the generator output will trip the DC Motor of the MMG set. Part 2 incorrect – Plausible in that loss of UPS does impact Feedwater Level Control System. RFW Control System Panel Display Stations on Panel 2-9-5 is disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

B INCORRECT: Part 1 incorrect – See explanation A. Part 2 correct - See explanation D.

C INCORRECT: Part 1 correct – See explanation D. Part 2 incorrect - See explanation A.

D **CORRECT:** Part 1 correct – The 1001 and 1003 breakers from an MMG set will trip on overvoltage or under frequency at the output of the MMG. Part 2 correct – Per 2-AOI-57-4, Subsequent action 4.2[1], verify RFW Control System is maintaining Reactor Water Level. The RFW Control System continues to control system parameters according to water level setpoint.

KA Justification:

The KA is met because the question tests the Candidates' ability to predict the impacts of Over voltage on the Unit 2 Unit Preferred System MMG which is an uninterruptable power supply. Then, assess impact of loss of UPS on FWLC to determine correct actions in accordance with 2-AOI-57-4.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidate must predict impact of loss of UPS on FWLC to determine appropriate action to take.

Technical Reference(s): OPL171.102 Rev. 7 (Attach if not previously provided)

2-AOI-57-4 Rev. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.102 V.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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
263000 DC Electrical Distribution	Tier #	2	------
K6.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION : • Battery ventilation	K/A #	263000K6.02	
	Importance Rating	2.5	------

Proposed Question: **# 48**

Battery Rooms 1, 2, and 3 HVAC Systems are not operating properly.

Which ONE of the following completes the statements below?

The concern is that __**(1)**__.

Plant procedures direct the utilization of an __**(2)**__.

A. **(1)** lead-calcium batteries tend to release toxic gas into the atmosphere at temperatures above 90 °F
(2) Emergency Exhaust Fan **ONLY**

B. **(1)** the design limit for hydrogen concentration in the rooms may be reached during battery charging operations
(2) Emergency Exhaust Fan **ONLY**

C. **(1)** lead-calcium batteries tend to release toxic gas into the atmosphere at temperatures above 90 °F
(2) Emergency Exhaust Fan **AND/OR** Portable Temporary Ventilation Equipment

D. **(1)** the design limit for hydrogen concentration in the rooms may be reached during battery charging operations
(2) Emergency Exhaust Fan **AND/OR** Portable Temporary Ventilation Equipment

Proposed Answer: D

Explanation (Optional):

A INCORRECT: Lead-calcium batteries suffer degraded performance at high temperatures but do not release toxic gas as a result. (See Attached Excerpts) An Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. But, there is another option provided in addition to the Emergency Exhaust Fan; which is portable temporary ventilation provided in Section 8.15 of 0-OI-31.

B INCORRECT: (See Attached Excerpts) First part correct in that hydrogen buildup to explosive levels is the concern. Second part incorrect as detailed in 'A' above.

C INCORRECT: First part incorrect as detailed in 'A' above. (See Attached Excerpts) Second part is correct in that an Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. AND operation / placement of portable temporary ventilation provided in Section 8.15 of 0-OI-31.

D **CORRECT:** First part correct in that hydrogen buildup to explosive levels is the concern. Second part is correct in that an Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. AND operation / placement of portable temporary ventilation provided in Section 8.15 of 0-OI-31.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the impacts of a loss / malfunction of battery ventilation on the DC Electrical Distribution System.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 0-OI-31, Rev. 136 (Attach if not previously provided)

OPL171.037 Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 V.B.10 (As available)

Question Source: Bank # HLT 1006

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

264000 Emergency Generators (Diesel/Jet)

K5.05 (10CFR 55.41.5)

Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) :

- Paralleling A.C. power sources

	RO	SRO
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	264000K5.05	
Importance Rating	3.4	------

Proposed Question: **# 49**

Diesel Generator (D/G) 'A' is synchronized to 4KV Shutdown Board 'A'. The instrumentation readings for the D/G are as follows:

- Voltage = 4160 VAC
- Frequency = 60 Hz
- Current = 280 amps
- Vars = 2200 Kvars
- Watts = 2600 Kw

Which ONE of the following is the correct action to obtain a 0.8 lagging power factor?

Take the ______________________________ .

[REFERENCE PROVIDED]

A. Governor control switch to **RAISE**.

B. Governor control switch to **LOWER**.

C. Voltage Regulator control switch to **RAISE**.

D. Voltage Regulator control switch to **LOWER**.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The governor controls KW not KVAR. Candidate misunderstanding of governor controlling speed and real load or KW.

B INCORRECT: The governor controls KW not KVAR. Candidate misunderstanding of governor controlling speed and real load or KW.

C INCORRECT: Taking the voltage regulator control switch to raise will increase generator excitation and raise KVAR. This will place the generator operating point farther away from the 0.8 power factor line. Candidate error in determining where the generator is operating in relationship to the 0.8 pf line.

D **CORRECT:** Need to lower KVARs by lowering generator excitation to lower reactive load. Desired operation at 2600 KW = a 1950 KVAR with a 0.8 lagging power factor.

KA Justification:

The KA is met because it tests knowledge of operational implications of paralleled AC sources design and how KW and KVAR are controlled to obtain optimum power factor on a DG

Question Cognitive Level:

This question is rated as C/A due to the requirement to solve a problem using references. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-82 Rev 112 (Attach if not previously provided)

OPL171.038 Rev17

Proposed references to be provided to applicants during examination: **0-OI-82 Illustration -1**

Learning Objective: V.B.1 (As available)

Question Source: Bank # LXR TEST OPL171.038 #3 Last used BFN 1006 Audit

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 **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air System (IAS)	Tier #	2	------
K6.07 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: • Valves	K/A #	300000K6.07	
	Importance Rating	2.5	------

Proposed Question: **# 50**

"G" Control Air Compressor's microcontroller fails, causing the Compressor Inlet Flow Valve to **throttle** open and the Compressor Bypass Control Valve to fail **fully** open.

Which ONE of following completes the statement below?

Control Air Header pressure will _______.

A. stabilize between 110 to 120 psig

B. stabilize between 90 to 110 psig

C. stabilize between 75 to 90 psig

D. increase to 132 psig

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** Plausible if the candidate doesn't know what the Bypass Control Valve does. IF he/she believes the valve bypasses the normal pressure control. 120 psig is a recognizable value in that it is the rated pressure of "G" Control Air Compressor..

B **CORRECT**: The two selected lead air compressors start at 98 psig; the first lag at 96 psig and the second lag at 94 psig. This would be sufficient to maintain control air header pressure between 90 and 110 psig

C INCORRECT: Plausible in that the compressor will run but not supply compressed air. Any air entering the compressor will be discharged through the Bypass Control Valve, to the Air Silencer, and back to atmosphere. Candidate may not understand that the Lead and Lag compressors will be able to maintain control air header pressure between approximately 90 to 110 psig

D INCORRECT: Compressor discharge pressure lowers. Plausible if the candidate doesn't know what the Bypass Control Valve does. IF he/she believes the valve bypasses the normal pressure control. Compressor Relief Valve setpoint is 132 psig.

KA Justification:

K/A asks for effect of a malfunction of a control air system valve. Question asks about the effect of failure of the Bypass Control Valve on the 'G' Air Compressor and Control Air System.

Question Cognitive Level:

Answering the question involves the multi-part mental process of assembling, sorting, or integrating the parts, which also requires the candidate to predict an outcome from the valves failure.

Technical Reference(s): OPL171.054, Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.9 (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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air System (IAS)	Tier #	2	------
K6.12 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM:	K/A #	300000K6.12	
• Breakers, relays and disconnects	Importance Rating	2.9	------

Proposed Question: **# 51**

Control Air Compressors 'A' **AND** 'C' are in service. A momentary loss of power to 480V Shutdown Board 1B occurs. Three seconds later, normal voltage is restored.

Which ONE of the following completes the statement below?

Control Air Compressor __**(1)**__ will trip **AND** __**(2)**__ automatically re-start when normal voltage is restored.

A. **(1)** A
(2) will

B. **(1)** C
(2) will

C. **(1)** A
(2) will **NOT**

D. **(1)** C
(2) will **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: 'A' compressor is powered from 480V SD Bd 1B, and will therefore trip. The compressor will not auto start when normal voltage is restored. Plausible in that Control Air Compressor G does restart if voltage restored within 4 seconds.

B INCORRECT: 'C' is powered from 480v Common Bd 1, which is **not** affected by this event. Plausible in that candidates could confuse 480V SD Bd 1B which does supply A with 480 V Common Bd 1 which does not. If C power supply had been momentarily interrupted, the second part would NOT be true with voltage restored within 4 seconds.

C **CORRECT:** 'A' compressor is powered from 480V SD Bd 1B, which **is** affected by this event. It does **NOT** have auto restart capability for ≤ 4 sec power loss, like Control Air Compressor 'G'.

D INCORRECT: C is powered from 480v Common Bd 1, which is **not** affected by this event. The 'G' compressor power loss logic is set @ ≤ 4 seconds on a loss of 480V RMOV Bd 2A.

KA Justification:

The effect of a breaker failure resulting in momentary loss of 480V Shutdown Board 1B to the instrument air system (Control Air at BFN) agrees with the stated K/A. .

Question Cognitive Level:

This question is high comprehension because the examinee must evaluate the situation and predict the effect on the instrument/control air system. This involves a multi-part mental process of assembling, sorting, and integrating the parts of the system.

Technical Reference(s): OPL171.054 Rev 15 (Attach if not previously provided)

0-OI-32 Rev 127

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1 (As available)

Question Source: Bank #

Modified Bank # BFN 0801 #52 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0801

(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 **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:

400000 Component Cooling Water

A1.01 (10CFR 55.41.5)

Ability to predict and/or monitor changes in parameters associated with operating the COMPONENT COOLING WATER SYSTEM controls including:

- CCW flow rate

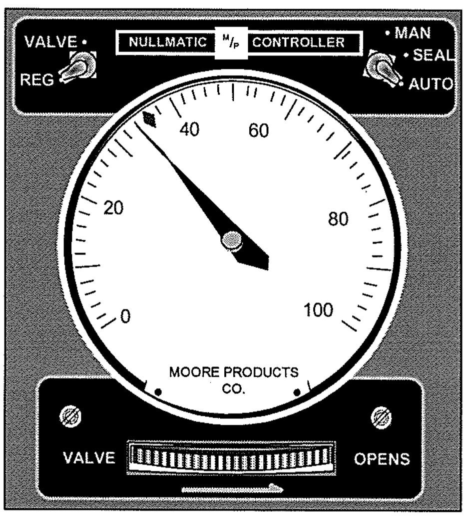
	RO	SRO
Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	400000A1.01	
Importance Rating	2.8	------

Proposed Question: **# 52**

Which ONE of the following completes the statement below?

The Unit 2 Reactor Building Closed Cooling Water (RBCCW) Temperature Controller, 2-TIC-24-80, is located in Unit 2 Reactor Building at __(1)__ .

If the controller is placed in AUTO with the indications shown below, the Temperature Control Valve will modulate to a more __(2)__ .



A. **(1)** Panel 2-25-196, Elevation 565'
(2) closed position

B. **(1)** RBCCW Heat Exchanger area, Elevation 593'
(2) close position

C. **(1)** Panel 2-25-196, Elevation 565'
(2) open position

D. **(1)** RBCCW Heat Exchanger area, Elevation 593'
(2) open position

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Part 1 correct - RBCCW Temp Controller, 2-TIC-24-80, is located in Unit 2 Reactor Building at Panel 2-25-196, Elevation 565'. Part 2 correct - with the RED indicator (Set Point) higher than the BLACK needle, indicates that actual temperature is cooler than desired. The TCV will modulate CLOSED.

B INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

D INCORRECT: Part 1 incorrect - Plausible in that several RCW valves associated with RBCCW are located at the RBCCW Heat Exchanger area, Reactor Building Elevation 593'. Part 2 incorrect - Plausibility based on misconception that with the feedback signal less than the control set point that the TCV would modulate Open to remove the deviation or that the controller is bypassing flow rather than controlling cooling water flow through the heat exchanger.

KA Justification:

The KA is met because the question tests the ability to predict and monitor changes in CCW Heat Exchanger flow in response to operating CCW Temperature control valve from Auto to Manual with a deviation between set point and feedback signal.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-24 Rev 77 (Attach if not previously provided)
OPL171.048 Rev 14

Proposed references to be provided to applicants during examination: ____

Learning Objective: ____ (As available)

Question Source: Bank #
Modified Bank # BFN 0801 #53 (Note changes or attach parent)
New

Question History: Last NRC Exam BFN 0801

(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 X

10 CFR Part 55 Content: 55.41 X
55.43

Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water	Tier #	2	------
A1.04 (10CFR 55.41.5)	Group #	1	------
Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including:	K/A #	400000A1.04	
• Surge Tank Level	Importance Rating	2.8	------

Proposed Question: **# 53**

Unit 2 RBCCW Heat Exchanger 2A is being filled and vented per 2-OI-70, "Reactor Building Closed Cooling Water System."

Which ONE of the following completes the statement below?

While filling the Heat Exchanger, RBCCW Surge Tank Level will lower until RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, __________.

A. is manually opened from the Control Room

B. is manually opened locally at the Surge Tank

C. automatically opens at 4 inches below the Surge Tank centerline

D. automatically opens at 4 inches above the Surge Tank centerline

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, is operated remotely from Control Room Panel 2-9-4.

B INCORRECT: Plausible in that manual BYPASS VLV, 2-FCV-70-1, is LOCALLY operated at the surge tank.

C INCORRECT: Plausible in that it is logical to have automatic make up capability to the RBCCW Surge Tank and 4 inches below the Surge Tank centerline is the set point for the Surge Tank Level Low Alarm. Additionally other plant head tanks automatically fill on low level. Examples: Demin Water Head Tank / PSC Surge Tank.

D INCORRECT: Plausible in that it is logical to have automatic make up capability to the RBCCW Surge Tank and 4 inches above the Surge Tank centerline is a recognizable value as the set point for the Surge Tank Level High Alarm. Additionally other plant head tanks automatically fill on low level. Examples: Demin Water Head Tank / PSC Surge Tank.

KA Justification:

The KA is met because the question tests candidates' ability to predict and monitor changes in Surge Tank Level associated with operating RBCCW controls to fill a Heat Exchanger.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 2-OI-70 Rev. 61 (Attach if not previously provided)

2-ARP-9-4C Rev. 30

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
201001 CRD Hydraulic	Tier #	2	
K5.05 (10CFR 55.41.5)	Group #	2	
Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM :	K/A #	201002K5.05	
• Indications of pump runout: Plant-Specific	Importance Rating	2.7	

Proposed Question: **# 54**

Unit 1 Control Rod Drive System has ruptured on the Charging Water Header upstream of the header restricting orifice.

Which ONE of the following completes the statement below?

This condition is indicated by the CRD Pump 1A motor amps being __**(1)**__ than normal **AND** the CRD Flow Control Valve traveling FULL __**(2)**__.

A. **(1)** LOWER
(2) OPEN

B. **(1)** HIGHER
(2) OPEN

C. **(1)** LOWER
(2) CLOSED

D. **(1)** HIGHER
(2) CLOSED

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 Correct – Plausibility based on misconception that pumping against backpressure of atmospheric as opposed to above Reactor Pressure would result in lower motor amps. Part 2 Correct – Plausible in that if CRD flow elements providing feedback to CRD FCV were downstream of where Charging Water Header ties in, the TCV would see low flow and go full open.

B INCORRECT: Part 1 Correct – See Explanation D. Part 2 Incorrect – See Explanation A.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

D **CORRECT**: Part 1 correct – The increase in Pump flow associated with going from normal flow to runout conditions would result in CRD Pump 1A motor amps higher than normal. Part 2 correct - CRD flow elements providing feedback to CRD FCV are upstream of where Charging Water Header ties in resulting in high flow sensed by the controller. The TCV would go full closed in response to the high flow condition.

KA Justification:

The KA is met because the question tests knowledge of indication and operational implications of CRD Pump 1A at runout due to a break in the system on the Charging Water Header.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.005 Rev. 17 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
201003 Control Rod and Drive Mechanism	Tier #	2	
K1.01 (10CFR 55.41.2 to 41.9)	Group #	2	
Knowledge of the physical connections and/or cause effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following:	K/A #	201003K1.01	
• Control rod drive hydraulic system	Importance Rating	3.2	

Proposed Question: **# 55**

During a UNIT 1 startup, a control rod drive mechanism is difficult to withdraw and remains at position 00.

The HCU hydraulic lines were vented.

Which ONE of the following completes the statement below in accordance with 1-OI-85, "Control Rod Drive System"?

GO TO _____.

A. ROD IN, **then** ROD OUT NOTCH with the CRD CONTROL SWITCH, release if rod moves

B. ROD OUT NOTCH with the CRD CONTROL SWITCH, **then** NOTCH OVERRIDE with the CRD NOTCH OVERRIDE SWITCH, release switches if rod moves

C. EMERGENCY IN with the CRD NOTCH OVERRIDE SWITCH, then simultaneously place the CRD CONTROL SWITCH in ROD OUT NOTCH, release switches if rod moves

D. EMERGENCY IN, **then** NOTCH OVERRIDE with the CRD NOTCH OVERRIDE SWITCH, **and then** simultaneously place CRD CONTROL SWITCH in ROD OUT NOTCH, release switches if rod moves

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This method may be used to vent some air from the CRDH lines but stem gives NOT believed to be air. RMCS settle time will be enforced between in and out signals. This method does give a withdrawal signal. Candidate may believe this will unstick the rod because it does give a withdrawal signal.

B INCORRECT: Would still ONLY get a single rod out notch signal. IF rod wouldn't move with single rod out notch signal, it won't move now. IF went to notch override first, then rod out, at least you would get a continuous withdrawal signal and vent any air from the withdrawal header/lines. Candidate misconception that notch override is giving a signal continuous withdrawal signal in this condition.

C INCORRECT: Drives rod in ONLY. Rod won't move out. It already has a continuous insert signal. May chose because of rod out notch signal. Candidate confusion that this is giving a continuous withdrawal signal.

D **CORRECT:** This is procedurally correct per 1-OI-85. The double clutch method is described.

KA Justification:

Question asks if relationship is understood between CRDH and CRDM and control room controls. It tests knowledge of double clutching a stuck rod to get it unstuck from the full in position. Candidate must understand RMCS, CRDH, and CRDM systems to determine how controls may be operated to unstick the control rod.

Question Cognitive Level:

This question has high cognitive value because; the candidate must recognize interaction between systems, including consequences and implications.

Technical Reference(s): 1-OI-85 Rev 23 (Attach if not previously provided)

OPL171.005 Rev 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.26 (As available)

Question Source: Bank # FERMI 2

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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

215001 Traversing In-core Probe

K4.01 (10CFR 55.41.7)

Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following:

- Primary containment isolation: Mark-I&II(Not-BWR1)

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	215001K4.01	
Importance Rating	3.4	------

Proposed Question: **# 56**

Unit 1 is operating at 100% Reactor Power with the "A" Traversing In-Core Probe (TIP) inserted in the core. A transient occurs resulting in the following plant conditions:

- Reactor Level is (-) 20 inches
- Drywell pressure is 1.5 psig

Which ONE of the following completes the statement below?

The "A" TIP will withdraw to the __**(1)**__ position **AND** the Ball Valve position will be __**(2)**__.

A. **(1)** 'PARKED'
(2) open

B. **(1)** 'PARKED'
(2) closed

C. **(1)** 'IN-SHIELD'
(2) open

D. **(1)** 'IN-SHIELD'
(2) closed

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - The TIP is withdrawn to the 'in-shield'. For the ball valve to close, it must be in the 'in-shield' position. Plausible in that there are TIP interlocks associated with the 'PARKED' position. Part 2 incorrect, the Ball Valve will close. Plausible in that shear valve will not close.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D. .

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT:** Per 1-AOI-64-2E, on a Group 8 signal, an AUTO withdraw signal is actuated. The TIP is withdrawn to the 'in-shield' position. Part 2 = Once in the 'in shield position, the Ball Valve will automatically close

KA Justification:

The KA is met because the question tests knowledge of TIP design feature and interlocks which provide for Primary containment isolation.

Question Cognitive Level:

Candidate must recognize Reactor Level is less than the set point for a Group 8 isolation and predict the impact on the TIP System.

Technical Reference(s): OPL171.17 Rev 15, OPL171.023 Rev 6 (Attach if not previously provided)

1-AOI-64-2E Rev 1 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.023 V.B.5 (As available)

Question Source: Bank # Hatch 09 #12

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Hatch 2009

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
230000 RHR/LPCI: Torus/Suppression Pool Cooling Mode	Tier #	2	
G2.4.31 (10CFR 55.41.10)	Group #	2	
Knowledge of annunciator alarms, indications, or response procedures.	K/A #	219000G2.4.31	
	Importance Rating	4.2	

Proposed Question: **# 57**

Unit 2 is at 100% Reactor Power with Residual Heat Removal (RHR) Loop II in Suppression Pool Cooling mode. The following alarms are received on **Unit 1**:

- DRYWELL PRESSURE HIGH HALF SCRAM, (1-9-4A, Window 8)
- RX PRESS LOW CORE SPRAY/RHR PERMISSIVE, (1-9-3C, Window 35)

Which ONE of the following describes the current status of **Unit 2** RHR system **AND** what actions, if any, must be taken to restore Suppression Pool Cooling on Unit 2?

A. **ALL** four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling IMMEDIATELY.

B. 2A **AND** 2C RHR Pumps are tripped. 2B **AND** 2D pumps are unaffected. **NO** additional action is required.

C. **ALL** four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling after a 60-second time delay.

D. 2B **AND** 2D RHR Pumps are tripped. 2A **AND** 2C pumps are unaffected. Place RHR Loop I in Suppression Pool Cooling IMMEDIATELY.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: This is plausible because all four RHR pumps on Unit 2 will trip, but they are locked out from manual start for 60 seconds based on Diesel Generator and/or Shutdown Board loading concerns.

B INCORRECT: This is plausible based on RHR Loop II being the preferred pumps for Unit 2.

C **CORRECT:** Candidate must determine that the combination of Unit 1 annunciators indicates a CAS initiation and the response of Unit 2 RHR pumps in Suppression Pool Cooling. Then, must recognize that Preferred and Non-preferred Emergency Core Cooling System (ECCS) Pumps do NOT apply with the given conditions. Unit 1 Preferred RHR pumps are 1A and 1C. Unit 2 Preferred RHR pumps are 2B and 2D. LOCA signals are divided into two separate signals, one referred to as a Pre Accident Signal (PAS) and the other referred to as a Common Accident Signal (CAS). If a unit receives a CAS, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards. All RHR and Core Spray pumps on the non-affected unit will trip (if running) and will be blocked from manual starting for 60 seconds. After 60 seconds all RHR pumps on the non-affected unit may be manually started.

D INCORRECT: This is plausible if taken from the perspective of Unit 1 operation, NOT Unit 2 operation.

KA Justification:

This question satisfies the KIA statement by requiring the candidate to use knowledge of annunciators for specific plant conditions to determine which RHR pumps can be used for Suppression Pool Cooling.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-ARP-9-3C Rev. 22 / OPL171.044 R. 17 (Attach if not previously provided)
1-ARP-9-4A Rev. 18 / 2-OI-74 Rev. 152

Proposed references to be provided to applicants during examination:

Learning Objective: OPL171.044 V.B.9/13 (As available)

Question Source: Bank # BFN 0610 #32
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 0610

(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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments: Question stem has been modified from original to meet KA. However, changes do not meet requirement of significantly modified question and is therefore identified as a Bank Question. Original attached.

Examination Outline Cross-reference:	Level	RO	SRO
230000 RHR/LPCI: Torus/Pool Spray Mode	Tier #	2	
K2.02 (10CFR 55.41.7)	Group #	2	
Knowledge of electrical power supplies to the following: • Pumps	K/A #	230000K2.02	
	Importance Rating	2.8	

Proposed Question: **# 58**

Unit 3 is operating at 100% Reactor Power with the Alternate Supply Breaker 1528 to 4 kV Unit Board 3B tagged out of service. An accident results in the following conditions:

- Unit Station Service Transformer 3B locks out
- Suppression Chamber Pressure reaches 3 psig
- 3A **AND** 3B RHR pumps are running in Suppression Chamber Spray Mode.

Which ONE of the following completes the statement below?

The power supply for the 4 kV Shutdown Board to RHR Pump 3A is __(1)__ **AND** RHR Pump 3B is__(2)__ .

A. **(1)** Common Station Service Transformer A
(2) Common Station Service Transformer A

B. **(1)** Common Station Service Transformer A
(2) its associated Emergency Diesel Generator

C. **(1)** its associated Emergency Diesel Generator
(2) Common Station Service Transformer A

D. **(1)** its associated Emergency Diesel Generator
(2) its associated Emergency Diesel Generator

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect – See Explanation C.

B **CORRECT**: 500 kV through USSTs is the normal supply to all U3 Unit Boards which in turn supply the 4kV Shutdown Boards. CSSTs are the alternate supply to the Unit Boards. EDGs are the emergency supply in case there is a loss of both normal and alternate supplies. Ordinarily the Unit Boards automatically transfer to alternate, however in this case the Unit Board 3B Alt is tagged out. So, when USST is lost, the 3C D/G will start and supply the 3EC 4 kV Shutdown Board which feeds RHR Pump 3B. Unit Board 3A will transfer and be supplied power via the CSST A. Unit Board 3A feeds 4 kV Shutdown Board 3EA which feeds RHR Pump 3A.

C INCORRECT: Part 1 and 2 incorrect - Plausible since the examinee must know which Unit Boards Supply which Shutdown Boards then RHR Pumps to eliminate these distractors.

D INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct – See Explanation B.

KA Justification:

The KA is met because it tests knowledge of electric power supplies to RHR Pumps.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.044 Rev. 17 (Attach if not previously provided)
OPL171.036 Rev. 12
3-ARP-9-8B Rev. 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036 V.B.8 (As available)

Question Source: Bank #
Modified Bank # Hatch 09 #22 (Note changes or attach parent)
New

Question History: Last NRC Exam Hatch 2009

(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 **X**

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
234000 Fuel Handling Equipment	Tier #	2	
A4.02 (10CFR 55.41.7)	Group #	2	
Ability to manually operate and/or monitor in the control room: • Control rod drive system	K/A #	234000A4.02	
	Importance Rating	3.4	

Proposed Question: **# 59**

Given the following:

- Unit 1 is in Mode 5
- The Refuel Platform is over the Spent Fuel Pool
- The Reactor Mode Switch is in START & HOT STBY for testing

Which ONE of the following identifies when a rod block will occur?

A. When the Refuel Platform Fuel Grapple is lowered.

B. When a load is placed on the Refuel Platform Fuel Grapple.

C. When the Refuel Platform is driven near or over the core.

D. Immediately when the Refuel Platform moves toward the core.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if the Mode Switch was in Refuel and Platform near or over the core.

B INCORRECT: Plausible in that this is true if the service platform hoist is loaded.

C **CORRECT:** As the Refuel Platform is driven near the core with the Mode Switch in Startup, a rod block will occur.

D INCORRECT**:** The refuel platform can move towards the core but will be stopped when the platform starts to move over the core

KA Justification:

The KA is met because the question tests the ability to monitor Control Rod Drive system in the control room as it applies to Fuel Handling Equipment.

Question Cognitive Level:

This question is rated as Fundamental Knowledge

Technical Reference(s): 0-GOI-100-3A Rev. 53 (Attach if not previously provided)

OPL171.053 Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.053 V.B.5 (As available)

Question Source: Bank # Cooper 08 #59

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Cooper 2008

(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 **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
259001 Reactor Feedwater System	Tier #	2	------
K5.03 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : • Turbine operation: TDRFP's-Only	K/A #	259001K5.03	
	Importance Rating	2.8	------

Proposed Question: **# 60**

RFPT 1A OVERSPEED TEST TRIP LOCKOUT, 1-HS-3-109A, has been placed in the 'ELEC' position per 1-OI-3, "Reactor Feedwater System," Section 8.9, "RFPT Overspeed Trip Test," when RFPT 1A experiences an **ACTUAL** over-speed condition.

Which ONE of the following describes the **AUTOMATIC** response of RFPT 1A?

A. Trips as a result of the electrical trip solenoid.

B. Trips as a result of the mechanical trip mechanism.

C. Will **ONLY** trip when 1-HS-3-109A is restored to the 'NORM' position.

D. Will **ONLY** trip when ELECT OVERSPEED TEST BYP, 1-HS-3-0109B is released from 'TEST' to 'NORM'

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The electrical trip solenoid is bypassed.

B **CORRECT:** The test blocks the electrical device trip but leaves the mechanical trip system active.

C INCORRECT: Yes the RFPT will trip when restored to NORM; however, the mechanical trip system remains active even in ELEC.

D INCORRECT: Yes the RFPT will trip when restored to NORM; however, the mechanical trip system remains active even in ELEC.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the operational implications of Turbine operation as it applies to the Reactor Feedwater System.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.026 Rev. 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.026 V.B.5 (As available)

Question Source: Bank # BFN 1006 Audit #63

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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
271000 Offgas System	Tier #	2	
K3.02 (10CFR 55.41.5)	Group #	2	
Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following: • †Off-site radioactive release rate	K/A #	271000K3.02	
	Importance Rating	3.3	

Proposed Question: **# 61**

Unit 2 Offgas Post Treat Radiation Monitor, 2-RM-90-265A, has failed downscale.

Which ONE of the following completes the statements below?

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, reaches the High-High-High setpoint, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(1)**__ close.

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, fails downscale, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(2)**__ close.

A. **(1)** will
(2) will

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

C. **(1)** will
(2) will **NOT**

D. **(1)** will **NOT**
(2) will **NOT**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Parts 1 and 2 correct - OG POST TREATMENT RAD MONITOR DOWNSCALE (55-4C-32) alarms when signal is < 1 cps and sends a trip signal to the Off-Gas isolation logic. OG POST-TREATMENT OFF-GAS HI-HI-HI/INOP (55-4C-35) alarms at 6.2X105 cps sends a trip signal to the Off-Gas isolation logic. Off-Gas isolation is a two-out-of-two logic. Downscale, Hi-Hi-Hi or INOP on RM-90-265A AND Downscale, Hi-Hi-Hi or INOP on RM-90-266A will automatically isolate the Off-Gas system after a 5 second time delay. (FCV-66-28 closes).

B INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

D INCORRECT: Part 1 incorrect – Plausible in that two channels are required for an isolation signal to 2-FCV-66-28 to be generated. Some process radiation monitors do not combine downscale with high radiation to generate the trips signal. Example: this combination would not result in a actuation of trip logic for Rx Zone Rad Monitors. Part 2 incorrect – Plausibility based on the misconception that the downscale does not result in a trip condition which is true of some process rad monitors. Example: Downscale on MSL Rad Monitors does not result in actuation of associated trip logic.

KA Justification:

The KA is met because the question tests candidates' knowledge of the effect that a malfunction of the OFFGAS SYSTEM Post Treatment Radiation Monitor will have on Offgas Isolation Valve 2-FCV-66-28 and therefore Off-site radioactive release rate.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.033 Rev. 13 (Attach if not previously provided)
2-OI-90 Rev. 79

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 V.B.3 (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 X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
288000 Plant Ventilation Systems	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	2	------
Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: • Isolation/initiation signals	K/A #	288000A3.01	
	Importance Rating	3.8	------

Proposed Question: **# 62**

Given the following Control Room Emergency Ventilation (CREV) system conditions:

- CREV Train A was started to prove operability following maintenance on the charcoal trays using the STOP-AUTO-START switch on Panel 9-22.
- The SYSTEM PRIORITY SELECTOR SWITCH is selected for "TRAIN-B".

Which ONE of the following completes the statement below that describes the CREV system response should a valid CREV initiation signal be received?

CREV Train B would__**(1)**__ **AND** CREV Train A would __**(2)**__.

A. **(1)** initiate
(2) shutdown

B. **(1) NOT** initiate
(2) shutdown

C. **(1)** initiate
(2) NOT shutdown

D. **(1) NOT** initiate
(2) NOT shutdown

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT Part 1 correct – See explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect - Normally, when an auto initiation signal is received, the TRAIN selected for "secondary" begins its start sequence but will not finish if the Primary CREV train is running. This is sensed by looking at the ΔP across the HEPA filter. Since Train B was selected as the Primary CREV unit, the start sequence does not look at the ΔP. Part 2 incorrect - This would be correct if CREV Train A was started using the AUTO-INITIATE TEST switch, as would be the case during the periodic surveillance test.

C **CORRECT:** Part 1 correct - CREV Train B will initiate without a time delay since the CREV UNIT PRIMARY SELECTOR SWITCH is selected for "TRAIN-B". Part 2 correct - CREV will not automatically shutdown with a valid initiation signal present.

D INCORRECT: Part 1 incorrect – See explanation B. Part 2 correct – See Explanation C.

KA Justification:

The KA is met because the question tests the ability to monitor automatic operation of Control Room Emergency Ventilation including system initiation signals for the given conditions.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-31 Rev. 136 (Attach if not previously provided)

OPL171.067 Rev 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.2.g (As available)

Question Source: Bank # 0707 #38

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0707

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
290001 Secondary Containment	Tier #	2	------
A1.01 (10CFR 55.41.5)	Group #	2	------
Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including:	K/A #	290001A1.01	
• System lineups	Importance Rating	3.1	------

Proposed Question: **# 63**

On Unit 1, Standby Gas Treatment System (SGTS) A, Control Switch 1-HS-65-18A on Panel 1-9-25 has been placed in the pull-to-lock position.

Which one of the following conditions would still cause SGTS A to start?

A. Unit 2 drywell pressure rises to 2.5 psig.

B. Unit 3 SGTS A start pushbutton is depressed.

C. The local (SGTS Building) SGTS A start pushbutton is depressed.

D. SGT TRAIN "A" INBD ISOL TEST SIG Keylock switch (HS-65-48A) is placed in the TEST position.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not auto start on 2.5 psig. Plausible in that this condition will normally cause SGTS A to start.

B INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not start with the Unit 3 SGTS A Start Pushbutton. Plausibility based misconception that Unit Control Switch will not affect operation from Unit 3.

C **CORRECT:** With control switch in pulled-out (STOP) position, the blower can still be started locally.

D INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not auto start with SGT TRAIN "A" INBD ISOL TEST SIG Keylock switch (HS-65-48A) placed in the TEST. Plausible in that this condition will normally cause SGTS A to start.

KA Justification:

The KA is met because the question tests the candidate's ability to predict changes in the SGTS associated with operating the SGTS Control Switch.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. Candidate must be able to predict the effect of changing the Control Switch position from its normal line up on the operation of the system.

Technical Reference(s):	OPL171.018 Rev 10	(Attach if not previously provided)
	0-OI-65 Rev 53	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:	Bank #	OPL171.018 #13	
	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	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
290002 Reactor Vessel Internals	Tier #	2	------
A2.01 (10CFR 55.41.5)	Group #	2	------
Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	290002A2.01	
• LOCA	Importance Rating	3.7	------

Proposed Question: **# 64**

Which ONE of the following completes the statements below?

Jet Pumps are designed such that following a DBA LOCA, a re-floodable core volume **NO** lower than __**(1)**__ is assured.

Following a DBA LOCA with **ALL** ECCS available, Severe Accident Management Guidelines __**(2)**__ be required to be entered.

A. **(1)** (-)180 inches
(2) will

B. **(1)** (-)180 inches
(2) will **NOT**

C. **(1)** (-)215 inches
(2) will

D. **(1)** (-)215 inches
(2) will **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that (-) 180 inches is a recognizable value associated with Low Reactor Water Level accident conditions and criteria for adequate core cooling. This is the minimum zero injection water level limit. Part 2 incorrect – Plausible in that a severe accident has occurred in a DBA LOCA and candidate may have the misconception that under these conditions SAMG entry is required regardless of whether adequate core cooling is met or not.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT**: Part 1 correct - Jet Pumps are designed such that following a DBA LOCA a re-floodable core volume **NO** lower than two thirds core height is assured. Two thirds core height corresponds to (-) 215 inches. Part 2 correct - ECCS is designed such that adequate core cooling will be met following a LOCA, assuming the worst case single active component failure in the ECCS. With all ECCS available, adequate core cooling is assure. Therefore, SAMGs are not required to be entered.

KA Justification:

The KA is met because the question tests the candidates' ability to predict the impacts of a LOCA on the Reactor Vessel Internals and based on those predictions, use procedures to control or mitigate the consequences of those abnormal conditions or operations in that the candidate must utilize the applicable sections and steps of EOI-1, "RPV Control," and EOI-C1, "Alternate Level Control" to determine that these procedures will not be exited for the SAMGs based on current plant conditions and predicted impact.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.212, Rev. 4 (Attach if not previously provided)

OPL171.201 Rev. 7 / OPL171.002 Rev. 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.212 V.B.2 (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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
290003 Control Room HVAC	Tier #	2	
K6.01 (10CFR 55.41.7)	Group #	2	
Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC :	K/A #	290003K6.01	
• Electrical power	Importance Rating	2.7	

Proposed Question: **# 65**

Which ONE of the following combinations of electrical board losses would result in **BOTH** Control Room Emergency Ventilation Fans being de-energized? (Assume normal alignment)

A. 480V Shutdown Board 3B; 4kV Shutdown Board 3EA

B. 480V Shutdown Board 1B; 4kV Shutdown Board 3EA

C. 480V Shutdown Board 3B; 4kV Shutdown Board A

D. 480V Shutdown Board 1B; 4kV Shutdown Board A

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT Part 1 correct – See explanation C. Part 2 incorrect - 3EA Plausible because it is normal feed to 480 V Sd Board 3A which feeds 480 V RMOV Board 3A. Since the B Fan is supplied by 480 VAC RMOV Board 3B, easily confused that A Fan would be supplied by 3A.

B INCORRECT: Part 1 incorrect – Plausible since the A Fan is supplied by 480 VAC RMOV Board 1A, easily confused that B Fan would be supplied by 480 VAC RMOV Board 1B whose normal feeder is 480 V Shutdown Board 1B. Part 2 incorrect – See Explanation A.

C **CORRECT**: Correct since the power supplies are 480 VAC RMOV Board 3B for fan B and 480 VAC RMOV Board 1A for fan A which is supplied by 4KV Shutdown Board A

D INCORRECT: Part 1 incorrect and Part 2 correct as explained above.

KA Justification:

The KA is met because the question tests whether the candidate has knowledge of the effect that a loss or malfunction of Electrical power will have on Control Room Emergency Ventilation.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.067, Rev. 16 (Attach if not previously provided)

0-OI-31 Att 3 Rev. 133

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.067 V.B.2 (As available)

Question Source: Bank # BFN 2004-301 #42

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 2004-301

(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 **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

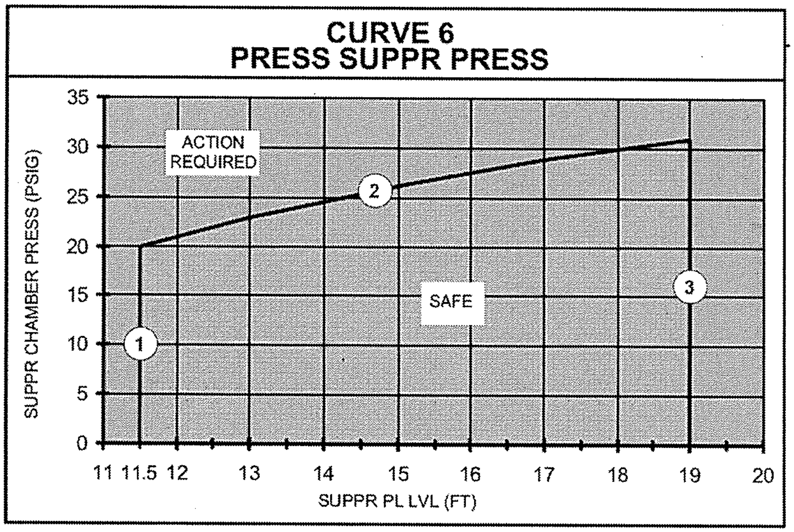
Examination Outline Cross-reference:

G2.1.25 (10CFR 55.41.10)

Ability to interpret reference materials, such as graphs, curves, tables, etc.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.25	
Importance Rating	3.9	------

Proposed Question: **# 66**



Which ONE of the following completes the statement below?

In accordance with the EOI Program Manual derivation, Line 1 on Curve 6, "Pressure Suppression Pressure," above, corresponds to the Suppression Pool Water Level at which the ________.

A. Downcomer Vents become uncovered

B. HPCI Turbine Exhaust opening becomes uncovered

C. Safety Relief Valve (SRV) Tailpipe openings become uncovered

D. Control Room Suppression Pool Water Narrow Range Level Indication goes off scale low

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (See attached excerpt) According to the EOI Program Manual, 11.5 feet (or Line 4) is the Suppression Pool Water Level which corresponds to the elevation of the downcomer vent openings.

B INCORRECT: The HPCI Turbine Exhaust becomes uncovered in the range of but above this value (at 12.75 feet) and is a significant direct Suppression Chamber Air Space pressurization event if HPCI remains running. PSP would be quickly exceeded.

C INCORRECT: SRV Tailpipes become uncovered around 5.5 feet. This is plausible because of the required ED at 11.5 feet. Normally, an ED on a parameter such as this is accomplished before you lose the ability to do so safely (within Safety Analyses assumptions).

D INCORRECT: Plausible because the X-Axis is based upon Suppression Pool Water Level and Narrow Range goes off-scale low at -25 inches which corresponds to approximately 13 feet.

KA Justification:

The KA is met because the question tests the candidate's ability to interpret Pressure Suppression Pressure Curve bounding limitations on Suppression Chamber Pressure versus Suppression Pool Level.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.201, Rev. 7 (Attach if not previously provided)

EOI Program Manual Sect. 2-VI-H, Rev. 10

0-TI-394, Rev. 4

Proposed references to be provided to applicants during examination: Embedded EOI Curve 6 - PSP

Learning Objective: OPL171.201 V.B.12 (As available)

Question Source: Bank # BFN 1006 #66

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55:43

Comments: This question was originally developed for an Audit Exam.

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.27 (CFR: 41.7)	Tier #	3	-------
Knowledge of system purpose and/or function.	Group #	-------	-------
	K/A #	G2.1.27	
	Importance Rating	3.9	-------

Proposed Question: **# 67**

Which ONE of the following is a Design Basis of HPCI?

A. Maintain sufficient reactor water inventory so the fuel won't overheat when a reactor isolation **AND** loss of feedwater occurs.

B. Make up water to the vessel in the event of a loss of coolant situation that does **NOT** result in rapid vessel depressurization.

C. Assures that the reactor core is adequately cooled to limit fuel clad temperature to < 1800 °F in the event of a large break in the reactor coolant system.

D. Assures that the reactor core is adequately cooled to limit primary containment pressure in the event of a small break in the reactor coolant system.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Maintains reactor water inventory so the fuel won't overheat is true, but this statement is the design basis for RCIC. Candidate may confuse the basis for HPCI and RCIC because they are similar in many respects. HPCI can also supply water to the reactor when a MSIV isolation and a loss of feedwater occur.

B **CORRECT:** Provides Adequate Core Cooling (ACC) for all break sizes that do NOT result in rapid depressurization of the reactor vessel. Correct design basis statement.

C INCORRECT: ECCS general design criteria is to limit fuel clad temperatures < 2200 °F. 1800 °F is EOI MZIRWL fuel clad temperature. Candidate may confuse EOI zero injection water level fuel clad temperature with ECCS design value.

D INCORRECT: HPCI design basis isn't about limiting primary containment pressure. Candidate may confuse primary containment design criteria with HPCI.

KA Justification:

The question meets the K/A by asking the design basis of HPCI.

Question Cognitive Level:

This is a low cognitive question. It asks for recall of the basis of the system or discrete bits of information.

Technical Reference(s): OPL171.042 Rev 20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1 (As available)

Question Source:	Bank #	Quad Cities 98	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Quad Cities 1998	

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.28 (10CFR 55.41.7)	Tier #	3	------
Knowledge of the purpose and function of major system components and controls.	Group #	------	------
	K/A #	G2.1.28	
	Importance Rating	4.1	------

Proposed Question: **# 68**

Which ONE of the following defines the purpose of the Rod Worth Minimizer (RWM) in accordance with Technical Specifications?

A. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when Reactor Power is < 10%.

B. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when Reactor Power is > 27%.

C. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when Reactor Power is < 10%.

D. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when Reactor Power is > 27%.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: The purpose of the RWM system is to limit control rod worth such that the fuel enthalpy limit of 280 cal/gm will not be exceeded during a Control Rod Drop Accident (CRDA). TS Table 3.3.2.1-1 requires the RWM to be operable in modes 1 and 2 with thermal power <10% RTP.

B INCORRECT: 1st part correct. 2nd part incorrect - Plausible in that ≥ 27% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

C INCORRECT: 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded due to additional rod withdrawal. 2nd part is correct.

D INCORRECT: 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded. 2nd part is incorrect. Plausible because ≥ 27% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

KA Justification:

The KA is met because the question tests knowledge of the purpose of the Rod Worth Minimizer.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.024 Rev. 14 (Attach if not previously provided)

TS 3.1-20 Amm 253

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.024 V.B.1 / 3 (As available)

Question Source: Bank # Hatch 09 #66

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Hatch 2009

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:

G2.2.2 (10CFR 55.41.6)
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.2	
Importance Rating	4.6	------

Proposed Question: **# 69**

Unit 1 Plant Startup is in progress.

Which ONE of the following completes the statement below?

In accordance with 1-GOI-100-1A,"Unit Startup," Control Rod withdrawal is limited to single notch when the __(1)__ SRM count rate doubling is reached **AND** must continue until __(2)__.

A. **(1)** fourth
(2) the Reactor is Critical

B. **(1)** fifth
(2) the Reactor is Critical

C. **(1)** fourth
(2) Reactor Power is in the heating range

D. **(1)** fifth
(2) Reactor Power is in the heating range

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect – Plausible in that Calculations have shown that when the initial SRM count rate has doubled 5 times that the reactor is very near criticality. Part 2 incorrect - Plausible in that 1-GOI-100-1A contains several cautions regarding the careful and controlled approach to criticality and the point of criticality is the trigger for several actions in the GOI.

C **CORRECT:** Part 1 correct – In accordance with 1-GOI-100-1A, A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. Part 2 correct – In accordance with 1-GOI-100-1A, once required, Control rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

KA Justification:

The KA is met because the question tests the candidates' ability to manipulate Control Rod console controls as required based on SRM response to operate the facility between shutdown and designated power levels.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 1-GOI-100-1A Rev. 23 (Attach if not previously provided)

OPL171.059 Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.059 V.B.3 / 4 (As available)

Question Source: Bank #

Modified Bank # Nine Mile 2 08 #70 (Note changes or attach parent)

New

Question History: Last NRC Exam Nine Mile 2 2008

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.39 (10CFR 55.41.7)	Tier #	3	------
Knowledge of less than or equal to one hour Technical Specification action statements for systems.	Group #	------	------
	K/A #	G2.2.39	
	Importance Rating		------

Proposed Question: **# 70**

Which ONE of the following completes the statement below?

When Reactor Steam Dome Pressure of __**(1)**__ is exceeded (as stated in Unit 2 Tech Spec 3.4.10, "Reactor Steam Dome Pressure"), it must be restored to within limits in a **MAXIMUM** completion time of __**(2)**__ .

A. **(1)** 1050 psig
(2) 15 minutes

B. **(1)** 1050 psig
(2) 1 hour

C. **(1)** 1073 psig
(2) 15 minutes

D. **(1)** 1073 psig
(2) 1 hour

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – In accordance with Unit 2 Tech Spec 3.4.10, the reactor steam dome pressure shall be ≤ 1050 psig. Part 2 correct – In accordance with Unit 2 Tech Spec 3.4.10 Condition A, if Reactor steam dome pressure not within limit, it must be restored with completion time of 15 minutes.

B INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

C INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

D INCORRECT: Part 1 is incorrect – Plausible in that this is a recognizable value associated with Reactor Pressure, i.e. EOI entry. Part 2 incorrect – Plausible in that 1 hour is common completion time in Tech Specs.

KA Justification:

The KA is met because the question tests knowledge of less than or equal to one hour Technical Specification action statements for TS 3.4.10, Reactor Steam Dome Pressure.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): U2 TS 3.4-30 Amm 254 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.43 (10CFR 55.41.10)	Tier #	3	------
Knowledge of the process used to track inoperable alarms.	Group #	------	------
	K/A #	G2.4.43	
	Importance Rating	3.0	------

Proposed Question: **# 71**

Which ONE of the following describes the meaning of a BLUE magnetic border being installed on a Main Control Room panel annunciator?

This type of border indicates that the annunciator ______.

A. has ONE **OR** more alarm inputs disabled

B. is "NOT ABNORMAL" for current plant conditions

C. is associated with ongoing testing **OR** maintenance

D. window is being relocated to a different window location

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with "Annunciator Disablement," OPDP-4, a blue magnetic border indicates that an alarm is out of service.

B INCORRECT: In accordance with "Annunciator System," 0-OI-55, a hot pink border indicates that an alarm is "NOT ABNORMAL" for current plant conditions.

C INCORRECT: In accordance with "Annunciator Disablement," OPDP-4, a white magnetic border indicates that an alarm is out of service for TESTING or MAINTENANCE.

D INCORRECT: In accordance with "Annunciator System," 0-OI-55, section 8.5, a yellow border is used to signify that an annunciator window is being relocated.

KA Justification:

The KA is met because the question tests knowledge of "Annunciator Disablement," OPDP-4, process for tracking inoperable alarms.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPDP-4 Rev. 4 (Attach if not previously provided)

0-OI-55 Rev. 46

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # BFN 1006 # 75 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.13 (10CFR 55.41.12) Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Tier #	3	------
	Group #	------	------
	K/A #	G2.3.13	
	Importance Rating	3.4	------

Proposed Question: **# 72**

A valve lineup is to be performed on valves with the following conditions:

- Area temperature is 115° F
- Area radiation is 40 mr/hr
- The valves are located 20' off the floor

Independent Verification of this valve lineup is expected to take 0.5 hour.

Which one of the following choices completes the statement below in accordance with NPG-SPP-10.3, "Verification Program?"

Based on the above conditions, Independent Verification of this lineup _______.

A. **CANNOT** be exempted

B. may be exempted due to elevation

C. may be exempted due to excessive dose

D. may be exempted due extreme temperature

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** Plausible in that candidate may believe dose levels are not high enough to warrant waiving IV. If the criteria for waiving IV was based on valve located in a High Radiation Area, this would be the correct answer.

B INCORRECT: Plausible in that there are multiple criteria in SPP-10.3 for waiving Independent Verification. However, valve in a hazardous location due to elevation is not

C **CORRECT:** Activities involving significant radiation exposure can be waived in accordance with SPP 10.3. As a guideline, an exposure greater than 10 mrem TEDE to perform verification would be considered excessive. This verification would result in dose of 20 mrem.

D INCORRECT: Plausible in that there are multiple criteria in SPP-10.3 for waiving Independent Verification. However, extreme temperature is not one.

KA Justification:

The KA is met because the question tests knowledge of radiological safety procedural requirements pertaining to licensed operator duties. Specifically, when the requirements for Independent Verification may be waived based on excessive dose.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome. Candidate must determine dose to be accumulated during the verification. Then, compare that to SPP-10.3 criteria for waiving IV to determine the correct answer.

Technical Reference(s): SPP-10.3 Rev. 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # Brunswick 08 # 72 (Note changes or attach parent)

New

Question History: Last NRC Exam Brunswick 2008

(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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.5 (10CFR 55.41.11/ 12)	Tier #	3	------
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Group #	------	------
	K/A #	G2.3.5	
	Importance Rating	2.9	------

Proposed Question: **# 73**

Which ONE of the following completes the statement below?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of __**(1)**__ ranges, **AND** has __**(2)**__ .

A. **(1)** TWO
(2) monitors in **ALL** three Units Control Rooms

B. **(1)** THREE
(2) monitors in **ALL** three Units Control Rooms

C. **(1)** TWO
(2) a monitor in Unit 2 Control Room **ONLY**

D. **(1)** THREE
(2) a monitor in Unit 2 Control Room **ONLY**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 = incorrect, Normal, Intermediate and high ranges are supplied. Part 2 = incorrect, The only remote monitoring is from Unit 2. Plausible in that Units 1 & 3 receive WRGRM alarms. 1/3-9-3A windows 6 & 13.

B INCORRECT: Part 1 = correct, Normal, Intermediate and high ranges are supplied. Part 2 = incorrect, The only remote monitoring is from Unit 2. Plausible in that Units 1 & 3 receive WRGRM alarms. 1/3-9-3A windows 6 & 13.

C INCORRECT: Part 1 = incorrect, Normal, Intermediate and high ranges are supplied. Part 2 = correct, Units 1 & 3 only receive common alarms. 1/3-9-3A windows 6 & 13. The only remote monitoring is from Unit 2.

D **CORRECT:** Part 1 = correct, Normal, Intermediate and high ranges are supplied. Part 2 = correct, Units 1 & 3 only receive common alarms. 1/3-9-3A windows 6 & 13. The only remote monitoring is from Unit 2.

KA Justification:

The KA is met because the question tests the ability to use the Wide Range Gaseous Effluent Radiation Monitor System which is a fixed radiation monitor.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.033 Rev 13 (Attach if not previously provided)

2-OI-90 Rev 79

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 V.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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.42 (10CFR 55.41.10)	Tier #	3	------
Knowledge of emergency response facilities.	Group #	------	------
	K/A #	G2.4.42	
	Importance Rating	2.6	------

Proposed Question: **# 74**

A plant emergency is in progress that requires a declaration in accordance with EPIP-1, "Emergency Plan Implementing Procedure."

The plant emergency in progress is **NOT** a security threat to facility protection.

Which one of the following is the "Lowest Classification" that the Operations Support Center (OSC) **AND** the Technical Support Center (TSC) must be activated?

	OSC	**TSC**
A.	Alert	Alert
B.	Alert	Site Area Emergency
C.	Site Area Emergency	Alert
D.	Site Area Emergency	Site Area Emergency

Proposed Answer: A

Explanation (Optional):

A **CORRECT**: Both parts correct - The TSC and OSC are required to be activated at the Alert or higher emergency classification

B INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

C INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

D INCORRECT: Both parts incorrect - Plausibility based on misconception that the OSC and TSC are not required to be activated until Site Area Emergency or higher.

KA Justification:

The KA is met because the question tests knowledge of what Emergency Action Level Emergency Response Facilities, OSC and TSC, are required to be activated.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): EPIP-6 Rev. 30 / EPIP-7 Rev. 27 (Attach if not previously provided)
OPL171.075 Rev. 25

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.075 V.B.10 (As available)

Question Source: Bank #
Modified Bank # Quad Cities 09 #75 (Note changes or attach parent)
New

Question History: Last NRC Exam Quad Cities 2009

(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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.47	Tier #	3	------
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	Group #	N/A	------
	K/A #	G2.4.47	
	Importance Rating	4.2	------

Proposed Question: **# 75**

ALL High Pressure Injection has been lost on Unit 2.

- At 16:00:00, Reactor Water Level is (-) 110 inches
- At 16:02:00, Reactor Water Level is (-) 118 inches
- Reactor Water Level continues to lower at the same rate

Which ONE of the following completes the statement below?

A Common Accident Signal will be initiated by __(1)__ Range level instruments **AND** the **EARLIEST** time that **ALL** Core Spray Pumps will have auto started is ___(2)___ .

A. (1) Emergency
(2) 16:03:07

B. (1) Post Accident
(2) 16:03:07

C. (1) Emergency
(2) 16:03:21

D. (1) Post Accident
(2) 16:03:21

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: (1) Incorrect, this instrument indicates (-)268 to (+)58 inches and initiates the Containment Spray Interlock. Candidate may select because instrument indication is within the desired range of Level 1. (2) Time is incorrect. Plausible in that this would be the correct answer for D/G Voltage Available (DGVA) sequence. Since there is no loss of offsite power, a Normal Voltage Available (NVA) sequence will occur.

C **CORRECT:** 1) Correct instrument. Emergency Range is (-)155 to (+)60 inches. Initiates HPCI, RCIC, RHR, CS and ADS. (2) Time is correct, level trend is 4 inches/min. Three minutes to Level 1, and with Normal Voltage Available (NVA), the last Core Spray Pump will sequence on 21 seconds after the accident signal is received.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

KA Justification:

The KA is met because the candidate must diagnose and determine trend and know correct control room instrument (range and function).

Question Cognitive Level:

This is higher cognitive because the examinee must know at what level Core Spray auto starts, calculate the time to the level, know the Core Spray sequence times based on the given plant conditions, and calculate the total time. The examinee must also know which type of instrumentation initiates the signal. He/she must use a multi-part mental process of assembling, sorting, or integrating parts of multiple systems to predict the outcome.

Technical Reference(s): OPL171.038 Rev 17 (Attach if not previously provided)

OPL171.003 Rev 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038 V.B.9, V.B.11 (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 X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295001 Partial or Complete Loss of Forced Core Flow Circulation	Tier #	------	1
AA2.02 (10CFR 55.43.5 - SRO Only)	Group #	------	1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :	K/A #	295001AA2.02	
• Neutron monitoring	Importance Rating	------	3.2

Proposed Question: **# 76**

Unit 1 was at 100% Reactor Power when Reactor Recirc Pump 1A tripped. Total Core Flow indication lowered to 50%.

Which ONE of the following completes the statements below?

Following the trip, APRM Flow Biased Scram set point will be__**(1)**__ Simulated Thermal Power.

The APRM Flow Biased Simulated Thermal Power – HIGH setpoint is required to be adjusted to Single Loop allowable value within a MAXIMUM of __**(2)**__ in accordance with T.S. 3.4.1, "Recirculation Loops Operating."

A. **(1)** 92%
(2) 12 hours

B. **(1)** 92%
(2) 24 hours

C. **(1)** 98%
(2) 12 hours

D. **(1)** 98%
(2) 24 hours

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect –Plausibility based on Flow biased setpoint for Control Rod Block is 0.66(w-Δw) +59%. .66(50-0) +59 = 92% STP. Part 2 incorrect - RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1. The 12 hour time is recognizable as the time required to place an Inop channel in trip per RPS Instrumentation TS.

B INCORRECT: Part 1 incorrect – See Explanation C. Part 2 incorrect - See Explanation C.

C INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct - See Explanation B.

D **CORRECT**: Part 1 correct - Flow biased setpoint for reactor scram is 0.66(w-Δw) + 65%. .66(50-0) +65 = 98% STP. Part 2 correct - RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1.

KA Justification:

The KA is met because the question tests the candidates' ability to determine and interpret APRM flow biased trip signals as they apply to a partial loss of forced core flow as a result of a trip of a Reactor Recirc Pump.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves application of Required Actions (Section 3) in accordance with rules of application requirements (Section 1). See Attached. Candidate must determine time requirement to apply APRM Flow Biased Simulated Thermal Power – HIGH setpoint as a result of the Recirc Pump Trip event.

Question Cognitive Level:

Question rated as C/A because Candidates' must use multi-part mental process in recognizing the effects of a Recirc Pump trip and core flow reduction to predict the change to the APRM flow biased set point.

Technical Reference(s): 1-AOI-68-1 Rev 3 (Attach if not previously provided)
OPL171.148 Rev 12
U1 TS 3.4-1/2 Amm 266 (Including version / revision number)
U1 TS B3.4-6 Rev. 45

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 V.B.2 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 0801 #91	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	BFN 0801	

(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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295005 Main Turbine Generator Trip / 3	Tier #	-------	1
G2.1.32 (10CFR 55.43.2 - SRO Only)	Group #	-------	1
Ability to explain and apply system limits and precautions.	K/A #	295005G2.1.32	
	Importance Rating	-------	4.0

Proposed Question: **# 77**

Which ONE of the following completes the statement below?

In accordance with the Unit 1 Bases for Tech Spec 3.3.1.1, "RPS Instrumentation," an RPS actuation is required as a result of Turbine Stop Valve Closure above a **MINIMUM** Reactor Power of __(1)__ to ensure the __(2)__ Safety Limit is not exceeded.

A. **(1)** 25%
(2) Reactor core MCPR

B. **(1)** 25%
(2) Reactor Coolant System RPV Pressure

C. **(1)** 30%
(2) Reactor core MCPR

D. **(1)** 30%
(2) Reactor Coolant System RPV Pressure

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that 25% is a recognizable value associated with Main Turbine instrumentation Tech Specs. The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at 25% RTP. Part 2 correct as detailed in 'C' below.

B INCORRECT: Part 1 incorrect as detailed in 'A' above. Part 2 is incorrect – Plausible in that Closure of the TSVs results in the loss of a heat sink that produces reactor pressure. However, ensuring safety limit for RPV Pressure is not exceeded is not the bases for the TSV RPS actuation.

C **CORRECT:** Part 1 correct – This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins. Part 2 correct – The Turbine Stop Valve - Closure Function is the primary scram signal for the turbine trip event. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

D INCORRECT: Part 1 correct as detailed in 'C' above. Part 2 incorrect as detailed in 'B' above.

KA Justification:

The KA is met because the question tests the candidates' ability to explain and apply limits associated with Main Turbine Generator Trip by asking the bases of RPS actuation in response to Turbine Control Valve closure and the Reactor Power limit for when the function is required.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO-only" Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves knowledge of TS bases for Turbine Stop Valve Closure. See attached Figure 1 flow chart.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): U1 TS B 3.3-23/24 Rev. 0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.028 V.B.9 (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295016 Control Room Abandonment	Tier #		1
G2.1.7 (10CFR 55.43.5 - SRO Only)	Group #		1
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	295016G2.1.7	
	Importance Rating		4.7

Proposed Question: **# 78**

Unit 3 was operating at 100% Reactor Power when the following occurred:

- Main Control Room evacuation is required due to a fire in the Control Bay
- The Backup Control Panel is manned twenty-five (25) minutes after evacuation of the Main Control Room
- The Unit Supervisor is informed that ONE SRV is continuously open **AND** a second SRV is cycling periodically

Which ONE of the following completes the statements below?

Based on the SRV status, Reactor Power is currently between __**(1)**__ .

In accordance with EPIP-1, "Emergency Plan Implementing Procedure," the **HIGHEST** emergency action level classification that is required for these conditions is a (an) __**(2)**__.

A. **(1)** 6% and 14%
(2) Alert

B. **(1)** 15% and 23%
(2) Alert

C. **(1)** 6% and 14%
(2) Site Area Emergency

D. **(1)** 15% and 23%
(2) Site Area Emergency

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect - With power greater than 15% two SRVs would be open continuously. Part 2 incorrect – Although the Backup Control Panel is manned within 25 minutes, the Alert is incorrect due to the inability to establish plant control within 20 minutes which includes controlling reactivity.

C **CORRECT**: Part 1 correct – each SRV will pass approximately 6.5% of total steam flow. With one SRV fully open and another cycling reactor power must be between the capacity of one and two relief valves. Part 2 correct - A Site Area Emergency must be declared due the inability to establish plant control within 20 minutes which includes controlling reactivity.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

KA Justification:

The KA is met because the question tests the candidates' ability to evaluate plant performance and make operational judgments. Based on SRV operation, candidate must conclude Reactor Power and make the required EAL Classification based on this evaluation of plant performance coupled with Control Room Abandonment.

SRO Only Justification:

This question meets the requirements of "Clarification Guidance for SRO-only Questions," Section II.F - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. [10 CFR 55.43(b)(6)] (See Attached). This question requires evaluating core conditions based on operating characteristics and determining emergency classifications based on core conditions coupled with Control Room Abandonment.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): EPIP-1 Rev. 46 / OPL171.009 Rev. 11 (Attach if not previously provided)
3-AOI-100-2 Rev. 20

Proposed references to be provided to applicants during examination:

Learning Objective: OPL171.075 V.B.2 (As available)

Question Source:	Bank #		
	Modified Bank #	Clinton 07 #90	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Clinton 2007	

(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	X
10 CFR Part 55 Content:	55.41	
	55.43	X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295021 Loss of Shutdown Cooling / 4	Tier #	------	1
G2.4.4 (10CFR 55.43.2 - SRO Only)	Group #	------	1
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	K/A #		295021G2.4.4
	Importance Rating	------	4.7

Proposed Question: **# 79**

Unit 1 RHR 1A is in Shutdown Cooling with Reactor Coolant temperature at 180° F. The Drywell Equipment Hatch is open. A leak on RHR Loop I results in the following:

- RHR LOOP I PUMP ROOM FLOOD LEVEL HIGH, (1-9-4C, Window 17), is in alarm
- RHR Loop I is secured **AND** isolated
- RHR Loop II is placed in service
- Reactor Coolant Temperature is now 215° F

Which ONE of the following completes the statements below?

Entry into 1-EOI-3, "Secondary Containment Control," __**(1)**__ required.

In accordance with EPIP-1, "Emergency Plan Implementing Procedure," __**(2)**__.

[REFERENCE PROVIDED]

A. **(1)** is
(2) Emergency Action Level for an Alert is met

B. **(1)** is
(2) Emergency Action Level for a Site Area Emergency is met

C. **(1)** is **NOT**
(2) Emergency Action Level for an Alert is met

D. **(1)** is **NOT**
(2) Emergency Action Level for a Site Area Emergency is met

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – RHR LOOP I PUMP ROOM FLOOD LEVEL HIGH alarm is indicative of Secondary Containment Area Water Level > 2" which is an EOI-3 entry condition. Part 2 correct – Reactor moderator temperature can NOT be maintained below 212° F and that with Primary Containment not maintained, Technical Specifications requires Mode 4 conditions, an Alert is required in accordance with EAL 1.5-A.

B INCORRECT: Part 1 correct – See explanation A. Part 2 incorrect – See Explanation D

C INCORRECT: Part 1 incorrect – See explanation D. Part 2 correct – See Explanation A.

D INCORRECT: Part 1 incorrect – Plausible in that not all alarms associated with degrading conditions occur at the EOI Entry level. Example – Drywell Pressure High alarms prior to the EOI entry level. Additionally, EOI-3 Entry is not required in Modes 4 and 5. The candidate must recognize that the event led to change to Mode 3 and therefore, EOI entry is required. Part 2 incorrect –Plausible in that a candidate may believe that the plant is not in the safe area of curve 1.5-S of EPIP-1.

KA Justification:

The KA is met because it tests ability to recognize abnormal indications (RHR LOOP I PUMP ROOM FLOOD LEVEL HIGH/ Loss of Shutdown Cooling / Reactor Coolant Temperature 215° F) for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. Entry levels met for EOI-3 and Loss of Decay Heat Removal EAL.

SRO Only Justification:

This question meets the requirements of "Clarification Guidance for SRO-only Questions," Section II.F - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. [10 CFR 55.43(b)(6)] (See Attached). Candidate must evaluate core conditions and determine emergency classifications based on core conditions. They must recognize Reactor moderator temperature can NOT be maintained below 212° F and with Primary Containment not maintained, Technical Specifications requires Mode 4 conditions. This results in declaration of an ALERT.

Question Cognitive Level:

Question rated as C/A because Candidates' must process multiple pieces of data including ECCS Room Flooded, elevated Reactor Coolant Temp, and Loss of S/D Cooling to ascertain EOI and EAL entry requirements.

Technical Reference(s): EPIP-1 Rev. 46 / U1 TS 3.6.1.1 Amm 234 (Attach if not previously provided)
1-9-4C Rev. 18 / OPL171.204 Rev. 7

Proposed references to be provided to applicants during examination: EPIP-1 EAL Matrix Section 1

Learning Objective: OPL171.075 V.B.2 (As available)
OPL171.204 V.B.2

Question Source: Bank #
Modified Bank # BFN 1006 #79 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 2010

(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 **X**

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295024 High Drywell Pressure	Tier #	----	
EA2.08 (10CFR 55.43.5 – SRO Only)	Group #	----	
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:	K/A #	295024EA2.08	
• Drywell radiation levels	Importance Rating	----	4.0

Proposed Question: **# 80**

Unit 3 was operating at 100% Reactor Power, when a leak in the Drywell resulted in the following conditions:

- Drywell Pressure is 57 psig and rising
- Suppression Chamber Pressure is 56 psig and rising
- Suppression Pool Level is 15 feet
- Drywell Radiation is 2500 R/Hr
- Reactor Water Level lowered to (-) 180 inches and is now (-) 170 inches and rising

Which ONE of the following identifies the required procedure to vent the Primary Containment **AND** the release rate requirements during the venting process in accordance with 3-EOI-2, "Primary Containment Control?"

A. 3-EOI-APPENDIX-12, "Primary Containment Venting"; vent irrespective of offsite release rates

B. 3-EOI-APPENDIX-12, "Primary Containment Venting" venting MUST be secured if approaching General Emergency Release Rate Limits

C. 3-EOI-APPENDIX-13,"Emergency Venting Primary Containment"; vent irrespective of offsite release rates

D. 3-EOI-APPENDIX-13,"Emergency Venting Primary Containment"; venting MUST be secured if approaching General Emergency Release Rate Limits

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if SC Pressure was not above 55 psig and Appendix 12 allowed venting irrespective of release rates.

B INCORRECT: Plausible in that this would be the correct answer if SC Pressure was not above 55 psig.

C **CORRECT:** In accordance with 3-EOI-2, "Primary Containment Control," with Suppression Chamber pressure 55 psig and Suppression Pool Level < 20 feet, venting of the Suppression Chamber is required irrespective of offsite release.

D INCORRECT: Plausible in that this would be the correct answer if venting was done IAW Appendix 12.

KA Justification:

The KA is met because the question tests the candidates' ability to interpret Drywell Radiation levels as they apply to High Drywell Pressure. Candidate must determine that Venting of the Suppression Chamber is required irrespective of release rate limits.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Candidate must assess plant conditions and then selecting a procedure, 3-EOI-APPENDIX-13, "Emergency Venting Primary Containment," due to high Suppression Chamber Pressure to mitigate the event. In making this selection candidate must further recognize that, venting is still required with the knowledge that Primary Containment Venting will result in release rates above ODCM limits.

Question Cognitive Level:

Question rated as C/A because it tests candidates' ability to process multiple pieces of data including Drywell/SC Pressure, Drywell Radiation Levels, Reactor Level and Suppression Pool Level to ascertain Venting requirements.

Technical Reference(s): 3-EOI-2 Rev 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.204 V.B.13 (As available)

Question Source:			
	Bank #	Hatch 09 #97	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Hatch 2009	

(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 **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295028 High Drywell Temperature	Tier #	------	3
EA2.02 (10CFR 55.43.5) – SRO ONLY	Group #	------	
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE :	K/A #	295028EA2.02	
• Reactor pressure	Importance Rating	------	3.9

Proposed Question: **# 81**

Given the following plant conditions on Unit 3:

- A steam line break has occurred inside the Drywell
- **ALL** Reactor Water Level (RWL) instruments display erratic indication
- Reactor Pressure **AND** Drywell Temperature are in the Action Required region of RPV Saturation Curve 8

Which ONE of the following completes the statement below?

The Unit Supervisor must select EOI flowchart __**(1)**__ for these conditions and raise injection to establish Reactor Pressure to a **MINIMUM** of __**(2)**__ above Suppression Chamber Pressure.

A. **(1)** 3-C-4, "RPV Flooding"
(2) 70 psig

B. **(1)** 3-C-2, "Emergency Depressurization"
(2) 70 psig

C. **(1)** 3-C-4, "RPV Flooding"
(2) 90 psig

D. **(1)** 3-C-2, "Emergency Depressurization"
(2) 90 psig

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – 3-EOI-3-C-4 is required because all level instruments are unavailable with Reactor Pressure and Drywell Temperature in the unsafe region of Curve 8 and erratic level instrument behavior. All actions associated with flooding and emergency depressurization are in 3-EOI-3-C-4. Part 2 correct – In accordance with 3-EOI-3-C-4, after Emergency Depressurizing, the crew must raise injection to establish Reactor Pressure to Minimum RPV Flooding Pressure of 70 psig above SC Pressure but as low as practicable.

B INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

D INCORRECT: Part 1 incorrect – Plausible because 3-EOI-3-C-2 is the normal emergency depressurization flowchart. Part 2 incorrect – Plausible because in accordance 1-EOI-1-C-4, the Minimum RPV Flooding Pressure for Unit 1 is 90 psig. Therefore, this would be the correct answer for Unit 1.

KA Justification:

The KA is met because the question tests ability to interpret Reactor Pressure as it applies to High Drywell Temperature. Candidate must recognize that with Drywell Temp / Reactor Pressure in the unsafe regions of the RPV Saturation curve and erratic level indications that all level indication is lost and then take appropriate actions in response.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Candidate must assess plant conditions and to determine that with Drywell Temp / Reactor Pressure in the unsafe regions of the RPV Saturation curve and erratic level indications that 3-C-4, "RPV Flooding" must be selected to Emergency Depressurize the Reactor to mitigate the event.

Question Cognitive Level:

The question is high cognitive because; solving it involves a multi-part mental process of assembling, sorting, or integrating the parts to solve a problem.

Technical Reference(s): 3-EOI-1 Rev 8 , 3-EOI-C-4 Rev 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:	Bank #		
	Modified Bank #	Hatch 09 #79	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Hatch 2009	

(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	X
10 CFR Part 55 Content:	55.41	
	55.43	X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-site Release Rate / 9	Tier #		1
G2.4.9 (10CFR 55.43.5 - SRO Only)	Group #		1
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	K/A #	295038G2.4.9	
	Importance Rating		4.2

Proposed Question: **# 82**

Unit 1 is at 100% Reactor Power:

- Main Steam Line radiation levels are greater than three times normal full power background
- OG AVG ANNUAL RELEASE RATE EXCEEDED 1-RA-90-157C, (1-9-4C, Window 27) is in alarm

Which ONE of the following completes the statement below?

The direction **AND** criteria to **CLOSE** MSIVs is contained in __**(1)**__ **AND** is based upon a determination that __**(2)**__.

A. **(1)** 0-EOI-4, "Radioactivity Release Control"
(2) releases are still in excess of Offsite Dose Calculation Manual limits

B. **(1)** Alarm Response Procedure 1-9-3A, Window 27 Section for MAIN STEAM LINE RADIATION HIGH-HIGH
(2) releases are still in excess of Offsite Dose Calculation Manual limits

C. **(1)** 0-EOI-4, "Radioactivity Release Control"
(2) the reactor will remain subcritical without boron under all conditions

D. **(1)** Alarm Response Procedure 1-9-3A, Window 27 Section for MAIN STEAM LINE RADIATION HIGH-HIGH
(2) the reactor will remain subcritical without boron under all conditions

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Both parts incorrect – as detailed in C and B below, respectively.

B INCORRECT: First part correct – as detailed in 'D' below. Second part incorrect – Plausible in that Main Steam Line Radiation High ARP contains action(s) associated with Offgas Radiation and ODCM limits.

C INCORRECT: First part incorrect – Plausible in that 0-EOI-4, does provide direction for isolating primary systems that are discharging into areas outside the primary and secondary containment. However, this step is not applicable under the specified conditions. Second part correct – as detailed in 'D' below.

D **CORRECT:** The Main Steam Line Rad Hi-Hi alarm, once validated, requires a core flow runback followed by a manual scram. Additionally, ARP specifies that if *not* in C-5 that MSIVs must be closed. If the reactor is shutdown under all conditions without boron, EOI Contingency C-5 will not be executed. Candidate must understand strategies associated with EOI/Contingency implementation.

KA Justification:

The KA is met because it tests candidate's knowledge of shutdown (ALL RODS IN) implications as they relate to excessive fuel failures inside the reactor core and the resultant high offsite release rates. As the ARP only specifies whether or not you are in "C-5,"additionally tests the candidate's knowledge of strategies associated with EOI and EOI Contingency implementation

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Candidate must determine whether or not C-5 requires execution for these conditions. The question requires assessing plant conditions to determine if MSIVs should be isolated and selecting the procedure to that provides this guidance to mitigate the event.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-ARP-9-4C, Rev. 18 (Attach if not previously provided)

1-ARP-9-3A, Rev. 40

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 V.B.14.a (As available)

Question Source: Bank #

Modified Bank # BFN 1006 #18 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:

295017 High Off-Site Release Rate

2.2.44 (10CFR 55.43.5 SRO – Only)

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	
K/A #	295017G2.2.44	
Importance Rating	------	4.4

Proposed Question: **# 83**

UNIT 2 was at 100% Reactor Power when an accident resulted in the following conditions:

- Main Steam Tunnel Temperature in the Turbine Building is 298 °F and rising.
- Main Steam Tunnel Temperature in the Reactor Building is 190 °F and rising.
- Main Steam Line C Inboard **AND** Outboard MSIVs can **NOT** be closed.
- Gaseous Release Rate Stack Noble Gas (WRGERMS) reading has been 6 x 10^{10} µCi/sec for 16 minutes.
- **NO** Offsite Emergency Response Facilities are operational.

Which ONE of the following completes the statements below?

In accordance with the EOIs, Emergency Depressurization __**(1)**__ required to be performed for these conditions..

The Shift Manager / Site Emergency Director __**(2)**__ delegate the determination of Protective Action Recommendation.

[REFERENCE PROVIDED]

A. **(1)** is
(2) can

B. **(1)** is **NOT**
(2) can

C. **(1)** is
(2) CANNOT

D. **(1)** is **NOT**
(2) CANNOT

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect – Plausible in that there are not 2 areas above their MAX SAFE limit. If candidate considers only EOI-3 requirements, this would be selected as correct. Part 2 incorrect – The Radiation Protection Manager is plausible in that his duties include assessment of site radiological conditions and recommendations for protective actions for onsite personnel.

C **CORRECT:** Part 1 correct – In accordance with 0-EOI-4, "Radioactive Release Control," if ED will reduce discharge outside of Primary and Secondary Containment and offsite radiation release is challenging General Emergency limit at the site boundary, ED is required. With failure of MSL C to isolate and temperature in the Turbine Building steam tunnel 298 °F and rising, there is indication of primary system discharging outside Primary and Secondary Containment. Part 2 correct – The Site Emergency Director must make any required recommendations (PARS) until the CECC is staffed. This responsibility cannot be delegated until CECC is in operation. Recommendations are required at General Emergency.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

KA Justification:

The KA is met because candidate must interpret control room indications for high area temperatures and MSIV position indications along with high offsite release data to verify the status of Primary Containment Isolation to determine correct operator actions and Radiological Emergency Plan actions.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Question requires detailed knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures based on interpretation of control room indications to verify that a leak is discharging outside of Primary and Secondary Containment. Also, determination of Protective Action Recommendations is a knowledge / ability unique to the SRO Position.

Question Cognitive Level:

This question is rated as C/A because it involves the multi-part mental process of assembling, sorting, or integrating the parts to solve the question posed in the stem.

Technical Reference(s): 0-EOI-4 Rev 5 / OPL171.075 Rev. 25 (Attach if not previously provided)
EPIP-5 Rev 39

Proposed references to be provided to applicants during examination: EPIP-1 Section 4

Learning Objective: OPL171.075 V.B.7 (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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Water Level / 5	Tier #	------	1
G2.4.47 (10CFR 55.43.5 - SRO Only)	Group #	------	2
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	K/A #	295029G2.4.47	
	Importance Rating	------	4.2

Proposed Question: **# 84**

A leak into Unit 2 Suppression Pool has resulted in the following indications:

- At 0200 Suppression Pool Level is (-) 3 inches and rising at 1 inch per hour

Which ONE of the following completes the statements below?

The Tech Spec Limit for 3.6.2.2, "Suppression Pool Level," will be reached at __**(1)**__.

The bases of the Tech Spec Suppression Pool upper level limit is to __**(2)**__ during a DBA LOCA.

A. **(1)** 0315
(2) ensure that peak primary containment pressure does not exceed maximum allowable values

B. **(1)** 0315
(2) prevent excessive clearing loads from S/RV discharges and excessive pool swell loads

C. **(1)** 0400
(2) ensure that peak primary containment pressure does not exceed maximum allowable values

D. **(1)** 0400
(2) prevent excessive clearing loads from S/RV discharges and excessive pool swell loads

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that the Suppression Chamber Water Level Abnormal will be received at this time due to high water level. Part 2 incorrect – Plausible in that this is a recognizable TS Bases associated with Suppression Pool Parameters. This is the bases of SP Temp limit.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT:** Part 1 correct – Tech Spec Limit of (-) 1 inch will be reached at 0400. Part 2 correct – This is the TS 3.6.2.2 Bases Suppression Pool upper level limit.

KA Justification:

The KA is met because the question tests candidates' ability to diagnose and recognize high Suppression Pool Water Level trend in an accurate and timely manner utilizing the appropriate control room reference material.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)] See Attached.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 TS 3.6-29 Am 253 (Attach if not previously provided)

U2 TS B 3.6-65 Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.016 V.B.12 (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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:

295032 High Secondary Containment Area Temperature

EA2.02 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE

- Equipment operability

	RO	SRO
Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #		295032EA2.02
Importance Rating	------	3.5

Proposed Question: **# 85**

Unit 3 was operating at 100% Reactor Power. RHR Pump 3B was tagged out for planned maintenance at 0600 on 1/13/11.

At 1000 on 1/14/11, a RCIC steam line leak occurred in the Reactor Building resulting in a trip of Loop I Core Spray Room Cooler.

Based on these conditions, which ONE of the following identifies the **LATEST** time that Unit 3 must be in Mode 3 in accordance with Tech Spec 3.5.1, "ECCS-Operating"?

[REFERENCE PROVIDED]

A. 2200 on 1/14/11

B. 2300 on 1/14/11

C. 1800 on 1/20/11

D. 2200 on 1/21/11

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if TS 3.0.3 required Mode 3 in 12 hours.

B **CORRECT**: With Core Spray 3A Pump Room Cooler inoperable, TRM 3.5.3 requires declaring Core Spray Loop I inoperable immediately. With Loop I CS and RHR Pump 3B INOP, TS 3.5.1 Condition H requires TS 3.0.3 Immediately. TS 3.0.3 requires Mode 3 in 13 hours.

C INCORRECT: Plausible in that this would be the correct answer if loss of the Core Spray Room Cooler did not require declaring the associated ECCS Pump inoperable.

D INCORRECT: Plausible in that this would be the correct answer if candidate believed that one inoperable RHR Pump does not result in the loop being considered inoperable and therefore not entering Condition A until the subsequent Core Spray 3A inoperability.

KA Justification:

The KA is met because the question tests ability to determine and/or interpret Equipment operability (Operability of ECCS Room Cooler and its impact on operability of CS System) as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE (A steam leak in RCIC Room resulting in trip of CS Room Cooler).

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves application of Required Actions (Section 3) in accordance with rules of application requirements (Section 1). See Attached. Candidate must determine that the CS Room Cooler is inoperable since it cannot maintain area temperature < 148° F and then determine that CS Loop I must also be declared inoperable. Then, they must apply the requirements of TS 3.5.1 and TS 3.0.3.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question and use reference to solve a problem.

Technical Reference(s): U3 TS 3.5-1 to 1a Amm 244 (Attach if not previously provided)
U3 TS 3.5-3 Amm 229
TRM 3.5.3 Rev. 0
U3 TS 3.5.1 Bases Rev. 0
U3 TS 3.0.3 Amm 226

Proposed references to be provided to applicants during examination: TS 3.5.1 No Bases

Learning Objective: OPL171.045 V.B.6 (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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
203000 RHR/LPCI: Injection Mode	Tier #	------	2
G2.2.4 (10CFR 55.43.5 - SRO Only)	Group #	------	1
(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.	K/A #		203000G2.2.4
	Importance Rating	------	3.6

Proposed Question: **# 86**

Unit 1 has experienced a Loss of Offsite Power concurrent with a LOCA. Multiple equipment failures have resulted in need for RHR Crosstie to be lined up for injection into the reactor.

Which ONE of the following completes the statements below?

Unit 1 RHR can be crosstied to Unit 2 RHR __**(1)**__ .

The Unit 2 RHR Pump Suction Valve interlocks must be defeated in accordance with __**(2)**__.

A. **(1)** Loop I
(2) 2-OI-74, "Residual Heat Removal System"

B. **(1)** Loop I
(2) 1-EOI Appendix 7C, "Alternate RPV Injection System Lineup RHR Crosstie"

C. **(1)** Loop II
(2) 2-OI-74, "Residual Heat Removal System"

D. **(1)** Loop II
(2) 1-EOI Appendix 7C, "Alternate RPV Injection System Lineup RHR Crosstie"

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect – See Explanation C.

B **CORRECT**: Part 1 correct – In accordance with 1-EOI Appendix 7C, Unit 1 RHR can be crosstied to Loop I Unit 2 RHR ONLY. Part 2 correct – RHR Pump Suction interlocks must be defeated to complete the crosstie and the instructions to defeat the interlocks is contained in 1-EOI Appendix 7C

C INCORRECT: Part 1 incorrect - Plausible in that this would be the correct answer if Unit 3 RHR is crosstied to Unit 2 RHR. Part 2 incorrect - Plausible in that defeating interlocks is sometimes directed in the associated Operating Instruction rather than the Appendix being performed. For example, when injecting CS per Appendix 6E with a loss of associated ECCS ATU Panel, defeating the reactor low pressure interlock would be performed in accordance with OI-75. Also, 2-OI-74 contains instructions for defeating various interlocks such as: Defeating the Rx Low Pressure Interlock on the RHR Loop 1/2 Injection and Inhibiting RHR Pump Auto Start and Auto Injection Logic

D INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct – See Explanation B.

KA Justification:

The KA is met because the question tests ability to explain variations in systems and procedures between units associated with RHR / LPCI Crosstie capabilities AND differences between Unit 1 and Unit 2 EOI Appendix 7C.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Unit Supervisor is required to analyze plant conditions and select the correct procedures to complete the required hardware modifications and to support the crosstie of Unit 1 and Unit 2 RHR.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-EOI Appendix 7C Rev. 1 (Attach if not previously provided)

OPL171.044 Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 V.B.3 (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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
262001 AC Electrical Distribution	Tier #	------	2
2.4.41 (10CFR 55.43.5 – SRO ONLY) Knowledge of the emergency action level thresholds and classifications.	Group #	------	1
	K/A #	262001G2.4.41	
	Importance Rating	------	4.6

Proposed Question: **# 87**

The following conditions exist on Unit 3:

- Reactor Power is 100%
- Emergency Diesel Generator 3EA is tagged out of service

The following sequence of events occur:

- 1130 **ALL** Offsite power is lost and NO Unit 3 EDG's tie to their associated Board
- 1140 EDG 3EB started and tied to its associated Board
- 1145 EDG 3EB Output Breaker trips open and cannot be closed
- 1155 EDG 3EC started and tied to its associated Board
- 1205 EDG 3EB Output Breaker is repaired and subsequently closed

Which ONE of the following identifies the **HIGHEST** emergency classification required **AND** who the Site Emergency Director should notify within five minutes of classifying the event?

[REFERENCE PROVIDED]

A. Alert;
Operations Duty Specialist

B. Alert;
State of Alabama

C. Site Area Emergency;
Operations Duty Specialist

D. Site Area Emergency;
State of Alabama

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – In accordance with EPIP-1, EAL 5.1-A1, Loss of voltage to any 3 unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes in Modes 1,2,or 3 and only one source of power to the remaining board requires declaration of an Alert. Part 2 correct - The Operations Duty Specialist (ODS) should be notified by the SM/SED within five minutes of the event classification..

B INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – Plausible in that Notification of the State of Alabama is required to be completed within 15 minutes from the time of emergency classification declaration.

C INCORRECT: Part 1 incorrect – Plausible in that this would be the correct answer in accordance with EPIP-1, EAL 5.1-S, Loss of voltage to ALL unit specific 4KV shutdown boards from Table 5.1 for greater than 15 minutes in Modes 1,2,or 3 requires declaration of a Site Area Emergency. Part 2 correct – See Explanation A

D INCORRECT: Part 1 incorrect – See Explanation C. Part 2 incorrect – See Explanation B.

KA Justification:

The KA is met because the question tests Emergency Action Level threshold and classification associated with AC Electrical Distribution with the loss of offsite power and subsequent Emergency Diesel Generator failures.

SRO Only Justification:

This question meets the requirements of "Clarification Guidance for SRO-only Questions," Section III. (See Attached). Classification of Emergencies is a knowledge / ability unique to the SRO position. Candidate must evaluate AC Electrical Distribution status and determine emergency classifications. This results in declaration of a Site Area Emergency.

Question Cognitive Level:

To solve the question the examinee must use a multi part mental process to assemble, sort, and integrate the parts of the plant conditions.

Technical Reference(s): EPIP-1, Rev 46 (Attach if not previously provided)

OPL171.075 Rev 25

Proposed references to be provided to applicants during examination: EPIP-1, Rev 46 Section 5

Learning Objective: V.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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments: The question has been modified from the original Brunswick 2008 #82 to be valid for Browns Ferry. However, it does not meet the requirements of a Significantly Modified Question so it is identified as a Bank Question. Original is attached.

Examination Outline Cross-reference:

261000 Standby Gas Treatment System

A2.12 (10CFR 55.43.5 - SRO Only)

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

- High fuel pool ventilation radiation: Plant-Specific

	RO	SRO
Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	261000A2.12	
Importance Rating	------	3.4

Proposed Question: **# 88**

Unit 3 is at 100% Reactor Power. Standby Gas Treatment System (SGTS) A was tagged out of service on 1/16/11 at 0600. SGTS B has been manually started. At 1000 on 1/16/11, a container is removed from the Unit 3 Spent Fuel Pool (SFP) resulting in the following Refuel Zone Radiation Monitor indications:

- 3-RM-90-140 Detector A is reading 73 mr/hr
- 3-RM-90-140 Detector B is reading 72 mr/hr
- 3-RM-90-141 Detector A is reading 71 mr/hr
- 3-RM-90-141 Detector B is reading 71 mr/hr

SGTS C did **NOT** start. The container was placed back in the SFP **AND** Refuel Zone Radiation Monitor indications returned to normal.

Which ONE of the following completes the statements below?

A Tech Spec required shutdown condition must be entered at __**(1)**__ in accordance with Tech Spec 3.6.4.3, "Standby Gas Treatment System."

A __**(2)**__ hour report to the NRC is required when the shutdown is commenced.

[REFERENCE PROVIDED]

A. **(1)** 1000 on 1/16/11
(2) four

B. **(1)** 0600 on 1/23/11
(2) four

C. **(1)** 1000 on 1/16/11
(2) one

D. **(1)** 0600 on 1/23/11
(2) one

Proposed Answer: **A**

Explanation (Optional):

A CORRECT: Part 1 correct - With Refuel Zone Radiation Monitor Channels A and D above the set point for automatic initiation of SGTS and the failure of SGTS C to start, SGTS C must be declared inoperable. With SGTS A and C inoperable, TS 3.6.4.3 Condition D requires immediate entry into TS 3.0.3. Part 2 correct – In accordance with SPP-3.5, "Regulatory Reporting Requirements," the initiation of any nuclear plant shutdown required by the plant's Technical Specifications requires a 4 hour NRC notification when the required shutdown is commenced.

B INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

D INCORRECT: Part 1 incorrect - Plausible in that if the right combination of channels for Automatic Start of SGTS did not exceed the set point, this would be the correct answer. SGTS C would still be operable so a shutdown condition would not be entered until SGTS A was tagged out for 7 days in accordance with TS 3.6.4.3 Conditions A and B. Part 2 incorrect – Plausible in that candidate may believe that reportability requirement is 1 hour or 8 hours.

KA Justification:

The KA is met because the question tests the candidates' ability to predict the impact of High fuel pool ventilation radiation on SGTS and with one train all ready out of service. Then, utilize Tech Specs and OPDP-8,"Limiting Conditions for Operation Tracking," to control the consequences of this abnormal condition.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves application of Required Actions (Section 3) in accordance with rules of application requirements (Section 1). See Attached. Candidate must determine that SGTS C is inoperable because it failed to start when the required number of channels reached the initiation set point. Then, they must determine when a TS shutdown condition is entered and reportability requirements. Determination of reportability requirements is also a function unique to the SRO position.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U3 TS 3.6-51/52 Amm 249 (Attach if not previously provided)
U3 TS 3.6-54 Amm 215
U3 TS 3.0-1 Amm 226
OPL171.033 Rev. 13 / SPP-3.5 Rev. 0

Proposed references to be provided to applicants during examination: U3 TS 3.6.4.3

Learning Objective: OPL171.033 V.B.5 (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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:

264000 EDGs

A2.09 (10CFR 55.43.5 – SRO Only)

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

- Loss of A.C. power

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	264000A2.09	
Importance Rating	------	4.1

Proposed Question: **# 89**

With Unit 1 Operating at 100% Reactor Power, a Loss of Offsite Power occurs.

Which ONE of the following completes the statements below?

In accordance with Tech Spec 3.8.1 Bases, “AC Sources – Operating,” on a Loss of Offsite Power, the **MAXIMUM** allowed time for Emergency Diesel Generators to energize their associated Shutdown Boards is __**(1)**__ seconds.

Direction to reset EECW to Control Air Compressors is contained in__**(2)**__ .

A. **(1)** 7
(2) 0-AOI-32-1, “Loss of Control and Service Air Compressors”

B. **(1)** 10
(2) 0-AOI-32-1, “Loss of Control and Service Air Compressors”

C. **(1)** 7
(2) 0-AOI-57-1A, “Loss of Offsite Power (161 and 500 KV)/Station Blackout”

D. **(1)** 10
(2) 0-AOI-57-1A, “Loss of Offsite Power (161 and 500 KV)/Station Blackout”

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that this is recognizable as the time second group of auto-connected loads is sequenced following D/G output breaker closing on DGVA sequencing. Part 2 incorrect – Plausible in Control and Service Air Compressors will be lost. However, guidance for resetting EECW supplies to Control Air Compressors is contained in 0-AOI-57-1A.

B INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

D **CORRECT**: Part 1 correct - Per TS 3.8.1, on an actual or simulated loss of offsite power signal the DG must auto-starts from standby condition and energize permanently connected loads in 10 seconds. Part 2 correct - Guidance for resetting EECW supplies to Control Air Compressors is contained in 0-AOI-57-1A.

KA Justification:

The KA is met because the question tests the candidates' ability to predict the impacts of Loss of A.C. power on the EMERGENCY GENERATORS and based on those predictions, use procedures to correct, control, or mitigate the consequences of the abnormal conditions including resetting EECW to Control Air Compressors.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E – Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.[10 CFR 55.43(b)(5)] The question requires assessing plant conditions and then selecting a procedure to mitigate or recover.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.038, Rev. 17 (Attach if not previously provided)

U1 TS 3.8.1 Am 235

U1 TS BASES 3.8.1 Rev. 52

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038 V.B.14 (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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air System (IAS)	Tier #	------	2
G2.2.36 (10CFR 55.43.2 - SRO Only)	Group #	------	1
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	K/A #	300000G2.2.36	
	Importance Rating	------	4.2

Proposed Question: **# 90**

Unit 3 is at 100% Reactor Power. Plant Control Air has been aligned to Drywell Control Air to allow maintenance on the Nitrogen Storage Tanks.

Which ONE of the following completes the statement below?

Technical Requirements Manual Section 3.6.3, "Drywell Control Air System," requires Reactor Thermal Power be reduced to less than or equal to __**(1)**__ power within __**(2)**__ if Plant Control Air is being used to supply the pneumatic control system inside primary containment.

A. **(1)** 15%
(2) 12 hours

B. **(1)** 15%
(2) 24 hours

C. **(1)** 25%
(2) 12 hours

D. **(1)** 25%
(2) 24 hours

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect See Explanation C.

B **CORRECT**: Part 1 and 2 correct - Technical Requirements Manual Section 3.6.3 requires reactor thermal power be reduced to less than or equal to 15% power within 24 hours if plant control air is being used to supply the pneumatic control system inside primary containment.

C INCORRECT: Part 1 and 2 incorrect - Plausible in that 25% Reactor Power and 12 hours are common power level / time requirements associated with Tech Spec Applicability and Surveillance Requirements. Example: SR 3.3.1.1.2 Not required to be performed until 12 hours after THERMAL POWER > 25% RTP.

D INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct See Explanation B.

KA Justification:

The KA is met because the question tests the candidates' ability to analyze the effect of maintenance activities on the status of limiting conditions for operations associated with the Control Air Systems.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves application of Required Actions (Section 3) in accordance with rules of application requirements (Section 1). See Attached. Candidate must determine power limitations and allowed time to achieve with Plant Control Air aligned to Drywell Control Air aligned to allow maintenance.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): TRM 3.6-5 Rev. 55 (Attach if not previously provided)
3-OI-32A Rev. 25 / OPL171.054 Rev. 15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.054 V.B.8 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:

202001 Recirculation

A2.13 (10CFR 55.43.5 - SRO Only)

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

- Carryunder

	RO	SRO
Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	202001A2.13	
Importance Rating	------	2.8

Proposed Question: **# 91**

Which ONE of the following completes the statements below?

Tech Spec 3.3.1.1, "Reactor Protection System (RPS) Instrumentation" **AND** its associated Bases for the Reactor Vessel Water Level - Low, Level 3 setpoint is to prevent significant carryunder __(1)__.

If this function is lost due to TWO inoperable channels in a trip system, then RPS trip capability must be restored __(2)__ .

A. **(1)** to ensure the accuracy of core D/P and level instrumentation
(2) Immediately

B. **(1)** to ensure the accuracy of core D/P and level instrumentation
(2) within 1 hour

C. **(1)** to protect available Reactor Recirc Pump Net Positive Suction Head
(2) Immediately

D. **(1)** to protect available Reactor Recirc Pump Net Positive Suction Head
(2) within 1 hour

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - Plausible in that Reactor Level instrumentation has taps in the Downcomer region where the dynamics are altered as a result of significant carryunder. Part 2 incorrect – Plausible in that Immediate is a common completion time in Tech Specs.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT**: Part 1 correct – In accordance with TS 3.3.1.1 Bases, The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder). Part 2 correct - In accordance with TS 3.3.1.1, Condition C, with one or more Functions with RPS trip capability not maintained, restore trip capability within 1 hour.

KA Justification:

The KA is met. To answer the question, the candidate must predict the impact of carryunder on the Recirculation System. Then, utilize Tech Specs and associated implementing procedures to mitigate the consequences loss of Level 3 RPS Channel designed to protect the Recirculation System from the impact of carryunder.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO-only" Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves knowledge of TS bases for Level 3 RPS.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Justification: Question requires knowledge of Tech Spec bases and is therefore, SRO-Only.

Technical Reference(s): U1 TS 3.3.1-2 Amm 262 (Attach if not previously provided)

U1 TS B 3.3.-18 Rev. 0 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.028 V.B.14 (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:

216000 Nuclear Boiler Instrumentation

G2.4.45 (10CFR 55.43.5)

Ability to prioritize and interpret the significance of each annunciator or alarm.

Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #		216000G2.4.45
Importance Rating	------	4.3

Proposed Question: **# 92**

The following alarms **AND** indications exist on Unit 3:

- DRYWELL PRESS HIGH, (3-9-3B, Window 23), is in alarm
- REACTOR VESSEL WTR LVL CH A LOW-LOW-LOW (3-9-5B, Window 4), is in alarm
- REACTOR VESSEL WTR LVL CH B LOW-LOW-LOW (3-9-5B, Window 5), is in alarm
- DRYWELL EQPT DR SUMP PUMP EXCESSIVE OPRN, (3-9-4B, Window 11), is in alarm
- Drywell Floor Drain Leakage is calculated at 100 gpm
- Group 1 PCIS Logic A Success light is **NOT** illuminated
- **ALL** other PCIS Logic Success lights are illuminated
- Dose equivalent Iodine-131 sample results indicate 16 μCi/gm

Which ONE of the following completes the statement below?

These alarms **AND** indications establish that _____.

A. a loss of the Fuel Clad Barrier **ONLY** exists

B. a loss of the Reactor Coolant System Barrier **ONLY** exists

C. a loss of the Reactor Coolant System Barrier **AND** Fuel Clad Barrier **ONLY** exists

D. a loss of the Containment Barrier **AND** Reactor Coolant System Barrier **ONLY** exists

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: plausible in that the threshold for fission product barrier loss - a Reactor coolant sample that yields a result of 300 μCi/gm Iodine-131 equivalent is indicative of cladding failure. 26 μCi/gm is a recognizable Tech Spec Number where action is required based on elevated I-131 levels. In addition RCS leakage is indicated.

B **CORRECT**: The threshold for Reactor Coolant System fission product barrier loss is considered to be consistent with Reactor coolant leakage of at least 50 GPM from the primary system.

C INCORRECT: The first part is correct, second part is plausible in that the threshold for fission product barrier loss – is a Reactor coolant sample that yields a result of 300 μCi/gm Iodine-131 equivalent is indicative of cladding failure.

D INCORRECT: The threshold for Primary Containment fission product barrier loss is considered to be consistent with the following: - Refer to 2.5-U. Unexplained Loss Of Containment Pressure / Exceeding SI-4.7.A.2.a Limits (Excessive N2 Makeup) / Inability To Isolate Any Line Exiting Containment When Isolation Is Required / Venting Irrespective Of Offsite Release Rates Per EOIs / SAMGs. Plausible in that REACTOR VESSEL WTR LVL CH A/B LOW-LOW-LOW alarms establish that MSIV isolation is required. Although the Group 1 PCIS Logic A Success light not illuminated indicates failure of the logic channel, one channel would meet the requirement to isolate the Main Steam Lines. The second part is correct.

KA Justification:

The KA is met because the question requires the candidate to interpret alarms and indications associated with the Nuclear Boiler System to determine Barrier losses in accordance with EPIP Bases.

SRO Only Justification:

This question meets the requirements of "Clarification Guidance for SRO-only Questions," Section II.F - Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. [10 CFR 55.43(b)(6)] (See Attached). This question requires evaluating core conditions, Reactor Coolant System Barrier and Containment Barrier in accordance with the Emergency Classification Procedure Technical Bases

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): EPIP-1 Rev. 46 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # BFN 1006 #100
PERRY 07 SRO #10 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006
Perry 2007

(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 X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:

271000 Offgas System

G2.2.40 (10CFR 55.43.2 - SRO Only)

Ability to apply Technical Specifications for a system.

	RO	SRO
Level	RO	SRO
Tier #	------	2
Group #	------	2
K/A #	271000G2.2.41	
Importance Rating	------	4.7

Proposed Question: **# 93**

Unit 3 is operating at 100% Reactor Power. Offgas Hydrogen Analyzer 3A was tagged out for planned maintenance at 0600 on 1/13/11.

At 0700 on 1/13/11, the Unit Supervisor discovers an error on Offgas Hydrogen Analyzer 3B Surveillance completed at 0400 on 1/13/11. Based on the corrected calculation, Offgas Hydrogen Analyzer 3B alarm setpoint is set too high to ensure the limit of TRM LCO 3.7.2 is not exceeded.

Which ONE of the following completes the statements below?

In accordance with TR 3.7.2, "Airborne Effluents," the concentration of hydrogen in Offgas downstream of the recombiners shall be limited to a **MAXIMUM** of __**(1)**__ . In accordance with TR 3.3.9, "Offgas Hydrogen Analyzer Instrumentation," Condition A must be entered with a start time of __**(2)**__ on 1/13/11.

[REFERENCE PROVIDED]

A. **(1)** 1%
   (2) 0600

B. **(1)** 1%
   (2) 0700

C. **(1)** 4%
   (2) 0600

D. **(1)** 4%
   (2) 0700

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect. Plausible in that this is the alarm set point for the Offgas H2 Analyzers. Part 2 correct – Plausibility based on misconception that start time should be when surveillance was complete.

B INCORRECT: Part 1 correct - See Explanation D. Part 2 incorrect - See Explanation A.

C INCORRECT: Part 1 incorrect - See Explanation A. Part 2 correct - See Explanation D.

D **CORRECT**: Part 1 correct - In accordance with TR 3.7.2, "Airborne Effluents," the concentration of hydrogen in Offgas downstream of the recombiners shall be limited to 4%. Part 2 correct - In accordance with TRM 3.0.2, start time is based on time of discovery

KA Justification:

The KA is met because the question tests the candidates' ability to apply Technical Specifications for the Offgas System

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question involves application of Required Actions (Section 3) in accordance with rules of application requirements (Section 1). See Attached. Candidate must determine the start time for Offgas Hydrogen Analyzers in accordance with LCO applicability section 3.0.2.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U3 TR 3.3-54 Rev. 16 (Attach if not previously provided)
U3 TRM 3.0-1 Rev. 44
U3 TRM 3.7-3 Rev. 0

Proposed references to be provided to applicants during examination: TR 3.3.9 (No SRs and No Bases)

Learning Objective: OPL171.087 V.B.10 (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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.3 (10CFR 55.43.2 – SRO Only)	Tier #	------	3
Knowledge of shift or short-term relief turnover practices.	Group #	------	------
	K/A #	G2.1.3	
	Importance Rating	------	3.9

Proposed Question: **# 94**

Which ONE of the following completes the statements below for Shift Turnover **AND** Control Board walk down requirements in accordance with OPDP-1,"Conduct of Operations?"

During shift turnover, the oncoming **Shift Manager** __(1)__ required to walk down the Control Boards with an off going RO or SRO.

The **Unit Supervisor** must walk down Main Control Room panels __(2)__.

A. **(1)** is
(2) once prior to mid shift brief **AND** once prior to end of shift turnover

B. **(1)** is **NOT**
(2) once prior to mid shift brief **AND** once prior to end of shift turnover

C. **(1)** is
(2) once every hour during power operations with a 25% grace period

D. **(1)** is **NOT**
(2) once every hour during power operations with a 25% grace period

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** Part 1 incorrect – See Explanation C. Part 2 correct - In accordance with OPDP-1, the Unit Supervisor walks down the main control room panels once each shift prior to the mid-shift brief and once prior to end-of-shift turnover.

B **CORRECT**: Part 1 correct – In accordance with OPDP-1, the oncoming Shift Manager is not required to conduct control board walk downs with an off-going Operator. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 incorrect – Plausible in that this would be the correct answer for the Unit Supervisor. Part 2 incorrect – See Explanation D.

D INCORRECT: Part 1 correct – See explanation B. Part 2 incorrect – Plausible in that this would be the correct answer for the Control Room Operators.

KA Justification:

The KA is met because the question tests the knowledge of shift relief turnover practices for Unit Supervisors.

SRO Only Justification:

This question is SRO Only because Unit Supervisor turnover and Control Room walk down requirements are knowledge / abilities unique to the SRO position.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): OPDP-1 Rev. 18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.071 V.B.16 (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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.4 (10CFR 55.43.2 – SRO Only)	Tier #	------	3
Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.	Group #	------	
	K/A #	G2.1.4	
	Importance Rating	------	3.8

Proposed Question: **# 95**

In accordance with OPDP-10, "License Status Maintenance, Reactivation and Proficiency for Non-Licensed Operators," which ONE of the following completes the statements for License Reactivation requirements?

Licensee requalification training must be verified current __**(1)**__ 40 hours of shift functions under instruction.

When **ALL** Reactivation requirements are met, the Licensed individual is authorized to resume licensed activities by the __**(2)**__.

A. **(1)** prior to standing
(2) Plant Manager

B. **(1)** prior to standing
(2) Site Licensing Manager

C. **(1)** after standing
(2) Plant Manager

D. **(1)** after standing
(2) Site Licensing Manager

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (1) correct, Licensee requalification training is current, including a simulator evaluation within the past 12 months in the position(s) to be assumed and the licensee has had a physical in the last two years. (To be verified prior to standing the 40 hours of shift functions under instruction.) (2) correct, Per OPDP-10 Appendix A:

The above licensed individual is authorized to resume licensed duties.
________________________ Date: _/_/_
Plant Manager

B INCORRECT: (1) correct, (2) incorrect, Plant Manager not Licensing Manager.

C INCORRECT: (1) incorrect, must be completed PRIOR to 40 hours. (2) correct,

D INCORRECT: Part 1 and 2 incorrect.

KA Justification:

The KA is met because the question tests the knowledge of individual licensed operator responsibilities associated with maintenance of active license status in accordance with 10CRF55.53.

SRO Only Justification:

This question meets the requirements of "Clarification Guidance for SRO-only Questions," Section II.A- Conditions and limitations in the facility license. [10 CFR 55.43(b)(1)]. The question deals with the requirement of OPDP-10 which is the implementing procedure for license maintenance of license status in accordance with 10CFR55.53

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): OPDP-10 rev 2 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # BFN 0801 #95

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0801

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:

G2.2.23 (10CFR 55.43.2 - SRO Only)

Ability to track Technical Specification limiting conditions for operations.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	
K/A #	G2.2.23	
Importance Rating	------	4.6

Proposed Question: **# 96**

Which ONE of the following completes the statements below?

If the criteria is met (in accordance with TS Section 1.3, "Completion Times") to apply a Completion Time extension, the total Completion Time allowed for completing a Required Action shall be limited to the __**(1)**__ restrictive of either:

- The stated Completion Time, as measured from the initial entry into the Condition, plus an additional __**(2)**__ ;**OR** the stated Completion Time as measured from discovery of the subsequent inoperability.

A. **(1)** more
(2) 12 hours

B. **(1)** less
(2) 12 hours

C. **(1)** more
(2) 24 hours

D. **(1)** less
(2) 24 hours

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct. Part 2 incorrect but plausible in that 12 hours is a common Tech Spec criteria / completion time.

B INCORRECT: Both are incorrect as explained below

C **CORRECT**: If the subsequent inoperability existed concurrent with the first inoperability and remained inoperable after the first inoperability was resolved, Completion Times may be extended in accordance with TS Section 1.3, "Completion Times". The completion time extension will be the more restrictive of initial entry plus an additional 24 hours or completion time as measured from discovery of the subsequent inoperability.

D INCORRECT: Part 1 incorrect but plausible in that when weighing alternative in accordance with Tech Spec use, application and applicability, the less restrictive is sometimes the criteria. Example: SR 3.0.3.

KA Justification:

The KA is met because the question tests the candidates' ability to track Technical Specification limiting conditions for operations by testing knowledge of Completion Time Extensions.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.B - Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]. The question tests knowledge of application of generic Limiting Condition for Operation (LCO) requirements (Section 1.3, Completion Times)

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): U1 TS 1.3-2 Amm 234 (Attach if not previously provided) (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087 V.B.10 (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.44 (10CFR 55.43.5 – SRO Only)	Tier #	----	3
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	Group #	----	
	K/A #	G2.2.44	
	Importance Rating	----	4.4

Proposed Question: **# 97**

A seismic event has resulted in the following Unit 2 plant conditions:

- **ALL** control rods are fully inserted
- RPV level is (-)150 inches and lowering slowly
- RPV pressure is 875 psig with a cooldown in progress at $\leq 90\ ^{O}F/hr$
- RHR Loop II is lined up for Drywell Spray
- **ALL** other ECCS systems are unavailable
- Drywell pressure is 4.8 psig and lowering
- ADS has been inhibited in accordance with 2-EOI-1, "RPV Control" step RC/L-7

Which ONE of the following describes the required actions to mitigate this event?

A. Enter 2-EOI-C1, "Alternate Level Control" and direct performance of 2-EOI-Appendix 6A, "Injection Subsystems Lineup Condensate."

B. Enter 2-EOI-C1, "Alternate Level Control" and direct performance of 2-EOI-Appendix 5A, "Injection System Lineup Condensate/Feedwater."

C. Enter 2-EOI-C2, "Emergency Depressurization" and direct performance of 2-EOI-Appendix 6A, "Injection Subsystems Lineup Condensate."

D. Enter 2-EOI-C2, "Emergency Depressurization" and direct performance of 2-EOI-Appendix 5A, "Injection System Lineup Condensate/Feedwater."

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – With level less than (-) 122 inches and lowering with no systems available to turn level for conditions, this is the appropriate leg of the EOIs to select. Part 2 correct – With the MSIVs closed and conditions not met to re-open MSIVs, this is the appropriate Appendix to select.

B INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect – See Explanation D.

C INCORRECT: Part 1 incorrect – See Explanation D. Part 2 correct – See Explanation A.

D INCORRECT: Part 1 incorrect. Direction to perform Emergency Depressurization based on reactor water level is given from EOI-C1 when RPV level drops below -162 inches. Other conditions given in the stem do not require Emergency Depressurization since Drywell Sprays have been initiated and appear to be effective. Part 2 incorrect. - Appendix 5A is a lineup for injection with RFPs which require MSIVs open. With RPV level below -122 inches, the MSIVs are closed. In addition, given all rods are in, performance of EOI Appendix 8A to bypass the MSIV low water level isolation is not appropriate

KA Justification:

The KA is met because the question tests the candidates' ability to interpret control room indications to verify the status of injection systems and understand how operator actions and directives affect plant and system conditions.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Candidate must assess plant conditions and then select a procedure, 2-EOI-APPENDIX 6A, "Injection System Lineup Condensate," due to MSIVs closed and conditions not met to re-open them.

Question Cognitive Level:

Question rated as C/A because it involves a multi-part mental process of assembling, sorting and integrating the plant conditions given to determine required section of EOIs and which Appendix to select.

Technical Reference(s): 2-EOI-1 Rev 12 / 2-EOI-2- C-1 Rev. 9 (Attach if not previously provided)
2-EOI Appendix 6A Rev. 4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.205 V.B.1 (As available)

Question Source: Bank # BFN 07 SRO #18
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 0707

(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 X

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

Examination Outline Cross-reference:

G2.3.12 (10CFR 55.43.4 – SRO Only)

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

	RO	SRO
Level		
Tier #	------	3
Group #	------	------
K/A #	G2.3.12	
Importance Rating	------	3.7

Proposed Question: **# 98**

Which ONE of the following completes the statements below in accordance with 1-GOI-200-2, "Primary Containment Initial Entry and Closeout?"

Initial Drywell Entry with the Reactor at Power must be approved by the __**(1)**__ .

A member of __**(2)**__ will remain at the Personnel Airlock in continuous communication with the Control Room **AND** with the persons in the Drywell.

A. **(1)** Shift Manager **ONLY**
   (2) Rad Protection

B. **(1)** Shift Manager **AND** Plant Manager
   (2) Rad Protection

C. **(1)** Shift Manager **ONLY**
   (2) Operations

D. **(1)** Shift Manager **AND** Plant Manager
   (2) Operations

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – This is plausible in that if the entry is made with the Reactor Mode switch is in SHUTDOWN or REFUEL position, Plant Manager authorization is not required and this would be the correct answer. Part 2 incorrect – Plausible in that Rad Protection has several responsibilities and communications requirements associated with Drywell Entry in accordance with 1-GOI-200-2.

B INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

D **CORRECT:** Part 1 correct – Initial entries are permitted only when the Reactor Mode switch is in SHUTDOWN, REFUEL, or STARTUP/HOT STANDBY position, unless drywell entry at power has been authorized by the Plant Manager. Shift Manager approval is required for all initial entries. Part 2 correct – In accordance with 1-GOI-200-2, if Primary Containment is required, a member of Operations will remain at the Personnel Airlock during drywell entry. This person will be in continuous communication with the Control Room and with the persons in the Drywell.

KA Justification:

The KA is met because the question tests knowledge of radiological safety principles pertaining to licensed operator duties associated with containment entry requirements

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.D - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)] The question tests knowledge of radiological safety requirements associated with Drywell Entry with the reactor at power.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): 1-GOI-200-2 Rev. 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: __________ (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1006 #98	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Browns Ferry 1006	

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

Examination Outline Cross-reference:

G2.3.7 (10CFR 55.43.4/5 – SRO Only)

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Level	RO	SRO
Tier #	------	3
Group #	------	
K/A #	G2.3.7	
Importance Rating	------	3.6

Proposed Question: **# 99**

In accordance with RCDP-3, "Administration of Radiation Work Permits," for normal and emergency situations, which ONE of the following completes the statements below?

During NORMAL situations, RADPRO Supervision __**(1)**__ authorize short term deviation from RWP requirements (for example, verbally requiring additional protective clothing), without revising the RWP.

If the Shift Manager authorizes IMMEDIATE entry into a High Radiation Area during emergency situations, then RADPRO escort __**(2)**__.

A. **(1)** may
(2) is still required

B. **(1)** may **NOT**
(2) is still required

C. **(1)** may **NOT**
(2) is **NOT** required

D. **(1)** may
(2) is **NOT** required

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 = correct – Per RCDP-3, "Administration of Radiation Work Permits", RADCON Supervision may authorize short term deviations (excluding regulatory and procedural deviations) from RWP requirements without revising the RWP. Part 2 = correct - Per RCDP-3, "Administration of Radiation Work Permits", in emergency situations where the Shift Manager authorizes immediate entry to an area, RADPRO is required to escort.

B INCORRECT: Part 1 = incorrect but plausible in that the candidate may assume that ALL RWP requirements need to be written within the RWP. Part 2 = correct for reasons detailed in A.

C INCORRECT: Part 1 = incorrect, as detailed in A. Part 2 = incorrect for reasons detailed in A and plausible in that the candidate may assume that since approval has been granted, only normal dosimetry is required w/o the need of an escort.

D INCORRECT: Part 1 = correct – Per RCDP-3, "Administration of Radiation Work Permits", RADCON Supervision may authorize short term deviations (excluding regulatory and procedural deviations) from RWP requirements without revising the RWP. Part 2 = incorrect for reasons detailed in A and plausible in that the candidate may assume that since approval has been granted, only normal dosimetry is required w/o the need of an escort.

KA Justification:

The KA is met because the question tests the ability to comply with radiation work permit requirements during normal or abnormal conditions.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.D - Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)] The question involves RWP requirements associated with radiation hazards.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): RCDP-3 Rev 2 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # BFN 0801 #99

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 09

(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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Examination Outline Cross-reference:

G2.4.22 (10CFR 55.43.5) – SRO ONLY

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

	RO	SRO
Level	RO	SRO
Tier #	------	3
Group #	------	
K/A #	G2.4.22	
Importance Rating	------	4.4

Proposed Question: **# 100**

With an ATWS, Emergency Operating Instructions (EOIs) require operators to reduce Recirc Pump speeds to minimum prior to tripping them if Reactor Power is above 5%.

Which ONE of the following identifies the **(1)** bases for this action **AND (2)** the EOI leg which requires it?

A. **(1)** To allow time for ARI to actuate thus allowing the Recirc Pumps to stay in operation for coolant circulation.
(2) C-5, Level / Power Control

B. **(1)** To allow time for ARI to actuate thus allowing the Recirc Pumps to stay in operation for coolant circulation.
(2) EOI-I, RPV Control, RC/Q leg

C. **(1)** To prevent tripping the turbine on high water level **AND** exceeding the capacity of the bypass valves.
(2) C-5, Level / Power Control

D. **(1)** To prevent tripping the turbine on high water level **AND** exceeding the capacity of the bypass valves.
(2) EOI-I, RPV Control, RC/Q leg

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - Plausible in that ARI is designed to dump air to HCU banks and SDV to atmosphere, ensuring rod insertion begins within 15 seconds and completes within 25 seconds. Therefore, the delay would provide time for ARI to complete the Scram, lowering power to less than 5% would possibly prevent need to trip Recirc Pumps. However, this is not the EOI Bases for this action. Part 2 incorrect – Plausible in that EOI-1 RC/L leg is exited and C-5 is entered with an ATWS and Reactor Power > 5%. However, the requirement to reduce Recirc to minimum prior to tripping is addressed in EOI-1 RC/Q leg.

B INCORRECT: Part 1 incorrect – See Explanation A. Part 2 correct – See Explanation D.

C INCORRECT: Part 1 correct – See Explanation D. Part 2 incorrect – See Explanation A.

D **CORRECT**: Part 1 correct – a recirculation flow runback is performed prior to tripping recirculation pumps in order to effect a more controlled reduction in reactor power. Even though the quickest reactor power reduction is achieved by tripping recirculation pumps, if a recirculation pump trip is initiated from a high reactor power level, the resulting plant transient may cause a main turbine trip due to rapid changes in steam flow, RPV pressure, and RPV water level. If reactor power is above turbine bypass valve capacity and the main turbine trips, RPV pressure will increase until one or more MSRVs open. Heatup of the suppression pool then begins.

KA Justification:

The KA is met because the question test knowledge of bases for prioritizing safety functions, i.e. Reactivity Control / CTMT Control with an ATWS condition present. The K/A requests knowledge of the bases for prioritizing safety functions during EOP operations and the question asks for the bases and emergency procedure.

SRO Only Justification:

This question is SRO Only because it meets the requirements of "Clarification Guidance for SRO," Section II.E - Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)] See Attached. Question involves knowledge of decision points in the EOIs that involve transitions to event specific contingency procedures.

Question Cognitive Level:

Question rated as Fundamental Knowledge.

Technical Reference(s): 1-EOI-1, Rev 0 / EOIPM 0-V-C Rev 1 (Attach if not previously provided)
OPL171.204 Rev 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:			
	Bank #		
	Modified Bank #	BFN 04 #98	(Note changes or attach parent)
	New		

Question History: Last NRC Exam Browns Ferry 2004

(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 **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments: