EXAMINATION ANSWER KEY

2017 RO NRC Exam

1 **ID: 1588335** **Points: 1.00**

An accident has occurred with the following:

- Conditions require ADS actuation.
- NO RHR pumps are available.

Which one of the following describes the minimum pump alignment required for Core Spray to satisfy the ADS permissive logic?

A. Any one Core Spray pump running satisfies the logic.

B. Any two Core Spray pumps running satisfy the logic.

C. At least two Core Spray pumps must be running to satisfy the logic, with either Core Spray pump A or B running, and either Core Spray pump C or D running.

D. At least two Core Spray pumps must be running to satisfy the logic, with either Core Spray pump A or C running, and either Core Spray pump B or D running.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	In order for Core Spray to satisfy the ADS permissive logic, at least two pumps must be running. One of the two pumps must be either A or B and one of the two pumps must be either C or D.
Distracters:	A	Two Core Spray pumps are required to be running. Plausible because only one RHR pump would be required.
	B	Some combinations of Core Spray pumps running would not satisfy the logic (A and B only, or C and D only). Plausible because two Core Spray pumps are required and no such restriction exists with RHR.
	D	The logic requires either A or B, and either C or D. Plausible because this is a similar arrangement with the pumps switched.

EXAMINATION ANSWER KEY

2017 RO NRC Exam

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588335
User-Defined ID:	
Cross Reference Number:	
Topic:	CS pump configuration needed for ADS
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item **X** Modified Bank (2015 NRC #2) ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC-227 D-4 and E-4	
Learning Objective:	PLOT-5001G 5b	
K/A System:	218000 ADS	Importance; RO 4.0
K/A Statement:	K1.02 - Knowledge of the physical connections and/or cause-effect relationships between ADS and the following: Low pressure core spray: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

2 **ID: 1588336** **Points: 1.00**

Unit 2 is operating at 50% power with the following:

- A fault occurs on the output of the 120 VAC UPS.
- Panel 20Y050, "Uninterruptible AC Power Distribution Panel" de-energizes.

Which one of the following describes the resulting status of APRM indications in the Control Room, in accordance with ON-112-2, Loss of Uninterruptible AC Power?

	C05 panel ODA indications	C37 panel APRM NUMAC indications
A.	Lost	Lost
B.	Lost	Available
C.	Available	Lost
D.	Available	Available

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	ODA indications are lost, but APRM NUMAC indications are still available.
Distracters:	A	APRM NUMAC indications are still available. Plausible because other APRM indications are lost.
	C	ODA indications are lost, but APRM NUMAC indications are still available. Plausible because this is the opposite of the actual answer and could be correct for some other malfunction that just affected NUMAC indications.
	D	ODA indications are lost. Plausible because these loads could be auctioneered with another power supply.

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588336
User-Defined ID:	
Cross Reference Number:	
Topic:	Effect of UPS loss on APRM indications
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ON-112-2	
Learning Objective:	PLOT-5058 5r	
K/A System:	262002 UPS (AC/DC)	Importance; RO 2.9
K/A Statement:	K1.19 - Knowledge of the physical connections and/or cause-effect relationships between UPS (AC/DC) and the following: Power range neutron monitoring system: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

3 **ID: 993643** **Points: 1.00**

The E124 load center has been inadvertently de-energized.

Which of the following valves will NOT operate due to this loss of power?

A. "A" RHR Pump Minimum Flow Valve (MO-2-10-16A)

B. "A" LPCI Inboard Injection Valve (MO-2-10-25A)

C. RCIC Minimum Flow Valve (MO-2-13-27)

D. HPCI CST Suction Valve (MO-2-23-17)

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	"A" RHR Min Flow Valve (MO-2-10-016A) is powered from E124.
Distracters:	B	LPCI Inbd Inj Valve (MO-25A) is powered from a normal seeking ABT (N210025), which is powered from E124/E324. Plausible because E124 does affect one of the two auctioneered power supplies.
	C	RCIC Min Flow Valve (MO-2-13-027) is powered from DC Supply 2DA. Plausible because this is a motor operated valve similar to the correct answer.
	D	HPCI CST Suction Valve (MO-2-23-017) is powered from DC supply 2DB. Plausible because this is a motor operated valve in another safety related system.

Question 3 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	1.00
System ID:	993643
User-Defined ID:	ILT-5010-2B-001
Cross Reference Number:	203000 K2.02
Topic:	ILT-5010-2b-001 The E124 load center has been inadvertently de-energized.
Num Field 1:	6490
Num Field 2:	A NRC
Text Field:	A

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(8)

Source Documentation

Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(993643)**	Previous NRC Exam Other Exam
Reference(s):	COL 10.1.A-2A	
Learning Objective:	PLOT-5010 2b	
K/A System:	203000 RHR/LPCI: Injection Mode	Importance; RO 2.5
K/A Statement:	K2.02 - Knowledge of electrical power supplies to the following: Valves	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

4 **ID: 993654** **Points: 1.00**

A fuse failure in 20D21, "125VDC Power Distribution Panel" causes a loss of Division I 125 VDC power to the Core Spray system only.

Which of the following describes the effect of this failure, if any, on the automatic response of Core Spray in the event of an accident?

A. All four Core Spray pumps would start and inject automatically.

B. Only the B Loop Core Spray pumps will start and inject automatically.

C. All four Core Spray pumps will start, but they will NOT automatically inject.

D. Only the B Loop Core Spray pumps will start, but they will NOT inject automatically.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	Div 1 125 VDC supplies power to the A Core Spray logic and Div II 125 VDC supplies power to the B Core Spray logic. The logic is arranged such that with the A logic not being powered, the A Core Spray pumps will not automatically start on a LOCA signal. The B Core Spray pumps will still automatically start and inject.
Distracters:	A	The A Core Spray pumps will not automatically start on a LOCA signal. Plausible because power is still available to half of the Core Spray logic and to the Core Spray pumps / valves.
	C	The A Core Spray pumps will not automatically start on a LOCA signal. The B Core Spray pumps will still automatically start and inject. Plausible because half the logic power is unavailable, such that this would be correct if both logics were required to start all pumps.
	D	The B Core Spray pumps will still automatically start and inject. Plausible if the logic or valve power was designed different.

Question 4 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	2.00
System ID:	993654
User-Defined ID:	ILT-5014-6D-001
Cross Reference Number:	209001 K2.03
Topic:	ILT-5014-6d-001 Division I 125 VDC power has been lost to the Core Spray system only.
Num Field 1:	5764
Num Field 2:	A NRC
Text Field:	A

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(993654)**	Previous NRC Exam Other Exam
Reference(s):	M-1-S-40	
Learning Objective:	PLOT5014 6d	
K/A System:	209001 LPCS	Importance; RO 2.9
K/A Statement:	K2.03 - Knowledge of electrical power supplies to the following: Initiation logic	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

5 **ID: 1588372** **Points: 1.00**

Unit 2 has experienced a scram with the following:

- HPCI automatically started and injected due to low Reactor water level.
- Reactor water level is now 0" and rising.
- HPCI is injecting 5000 gpm.
- A failure is preventing reset of the HPCI initiation signal.

Which one of the following describes the ability to control the HPCI injection flow rate and place HPCI in the CST to CST mode for Reactor pressure control, in accordance with SO-23.7.A-2, High Pressure Coolant Injection System Automatic Initiation Response?

HPCI ___(1)___ be controlled to inject at a lower flow rate. HPCI ___(2)___ be placed in the CST to CST mode for Reactor pressure control.

A. (1) can
(2) can

B. (1) can
(2) CANNOT

C. (1) CANNOT
(2) can

D. (1) CANNOT
(2) CANNOT

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	Failure of the HPCI initiation signal to reset does not prevent lowering the HPCI injection flow rate. Even with the initiation signal sealed in, the flow controller can be used to either lower the automatic flow rate setpoint or to manually lower flow rate. However, the inability to reset the initiation signal does prevent placing HPCI in the CST to CST mode for Reactor pressure control. The presence of the initiation signal keeps MO-24 (common test return to CST valve) and MO-21 (full flow test valve) closed. Both of these valves must be opened to initiate CST to CST mode.
Distracters:	A	The CST to CST mode cannot be accomplished. Plausible because MO-24 does have an auto close signal bypass switch, but no such feature exists for MO-21, so the lineup cannot be accomplished.
	C	Lowering HPCI injection flow rate is possible. Plausible because some emergency injection systems are designed such that while an injection signal is sealed in, flow rate cannot be lowered. The CST to CST mode cannot be accomplished. Plausible because MO-24 does have an auto close signal bypass switch, but no such feature exists for MO-21, so the lineup cannot be accomplished.
	D	Lowering HPCI injection flow rate is possible. Plausible because some emergency injection systems are designed such that while an injection signal is sealed in, flow rate cannot be lowered.

Question 5 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588372
User-Defined ID:	
Cross Reference Number:	
Topic:	CST to CST mode with initiation signal sealed in
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(8)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	SO 23.1.B-2, SO 23.7.A-2	
Learning Objective:	PLOT-5023 6b	
K/A System:	206000 HPCI	Importance; RO 3.8
K/A Statement:	K3.02 - Knowledge of the effect that a loss or malfunction of the HPCI will have on following: Reactor pressure control: BWR-2, 3, 4	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

6 **ID: 1588373** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- APRM 1 is bypassed.
- APRM 2 fails INOP.

Which one of the following describes the plant response?

This failure causes an...

A. alarm ONLY.

B. alarm and a rod block ONLY.

C. alarm, rod block, and half scram ONLY

D. alarm, rod block, and full scram.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	An APRM inop condition causes an alarm (211 A-3), a rod block, and a vote to the 2/4 voters. The bypassed APRM does affect the voter logic, but the voters need to have two votes to cause a scram signal. The bypassed APRM does not provide this vote. With only one vote from the INOP APRM, no scram signal is received.
Distracters:	A	A rod block occurs also. Plausible because there is no actual high power level (and corresponding APRM upscale condition) with the given failure.
	C	No scram signal is received. Plausible because a half scram would occur on other plants that do not have 2/4 voters.
	D	No scram signal is received. Plausible because 2 APRMs are in an off-normal state and 2 APRMs are enough to cause a full scram if they are both either INOP or HI-HI.

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588373
User-Defined ID:	
Cross Reference Number:	
Topic:	APRM Fails Inop with Bypassed APRM
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item **X** Modified Bank **(994392)** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	ARC 211 A-3	
Learning Objective:	PLOT-5060 6a	
K/A System:	215005 APRM / LPRM	Importance; RO 4.0
K/A Statement:	K3.01 - Knowledge of the effect that a loss or malfunction of the APRM / LPRM will have on following: RPS	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

7 **ID: 1588376** **Points: 1.00**

Unit 2 has experienced the following

- An Emergency Blowdown is in progress per T-112, "Emergency Blowdown".
- 5 SRV control switches are in OPEN.
- Torus pressure is 20 psig and stable.
- Drywell pressure is 22 psig and stable.
- Torus water level is 14 feet and stable.
- Reactor pressure is 35 psig.

Which one of the following describes the control room position indication and actual position for these 5 SRVs?

The SRVs indicate...

A. open and are actually open.

B. open, but are actually closed.

C. closed, but are actually open.

D. closed and are actually closed.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	With Reactor pressure less than 50 psig above Torus pressure, the SRVs are closed despite the position of their control switches. This is because Reactor pressure must be great enough to overcome spring pressure for the valves to be open. The position indication also indicates closed because it is based on acoustic monitors (actual flow) rather than switch position or solenoid position.
Distracters:	A	The SRVs are closed and indicate closed. Plausible because the control switches are in OPEN, and the valves would be open and indicate open if Reactor pressure were higher.
	B	The SRVs indicate closed. Plausible because the control switches are in OPEN, however position indicate is based on acoustic monitors (actual flow) rather than switch position or solenoid position.
	C	The SRVs are closed. Plausible because the control switches are in OPEN, and the valves would be open if Reactor pressure were higher. Also plausible if valve were held open mechanically/electrically but flow was too low to actuate acoustic monitors.

Question 7 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588376
User-Defined ID:	
Cross Reference Number:	239002K4.07
Topic:	SRV response at low pressure
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(3)

Source Documentation

Source:	New Exam item Modified Bank **X** ILT Exam Bank **(1588376)**	Previous NRC Exam Other Exam Bank
Reference(s):	T-112 Bases, P-S-31	
Learning Objective:	PLOT-5001A 3d	
K/A System:	239002 SRVs	Importance; RO 3.1
K/A Statement:	K4.07 - Knowledge of SRVs design feature(s) and/or interlocks which provide for the following: Minimum steam pressure required to keep SRV open or to open SRV	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:	TRH 9/5/16 – 2007 Cert #13 is a similar question.	

8 **ID: 1588377** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- The A and C Instrument Air Compressors are tripped.
- The A Instrument Air Header is at 85 psig.
- The B Instrument Air Header is at 105 psig.

Which one of the following describes the response of AO-80250D, Backup Air Control Valve, and the effect on the A Instrument Air Header?

AO-80250D ___(1)___. This action alone ___(2)___ restore the A Instrument Air Header pressure.

A. (1) automatically opens
(2) will

B. (1) automatically opens
(2) will NOT

C. (1) must be manually opened
(2) will

D. (1) must be manually opened
(2) will NOT

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	AO-80250D must be opened manually because to open automatically, both the A and B Instrument Air Header pressure must be <90 psig. The discharge is normally only aligned to the B header and requires manual valving to align to the A Header to restore pressure.
Distracters:	A	AO-80250D does not automatically open because the B Instrument Air Header pressure is not <90 psig. Plausible because this valve would automatically open if both air header pressures were <90 psig. AO-80250D is normally aligned to only the B Instrument Air Header, therefore additional action is required for it to restore the A Instrument Air Header pressure. Plausible if the normal alignment and system design provided a flow path to the A Instrument Air Header.
	B	AO-80250D does not automatically open because the B Instrument Air Header pressure is not <90 psig. Plausible because this valve would automatically open if both air header pressures were <90 psig.
	C	AO-80250D is normally aligned to only the B Instrument Air Header, therefore additional action is required for it to restore the A Instrument Air Header pressure. Plausible if the normal alignment and system design provided a flow path to the A Instrument Air Header.

Question 8 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588377
User-Defined ID:	
Cross Reference Number:	
Topic:	AO-80250D response to IA header A low pressure
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(4)

Source Documentation

Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(2008 Cert #52)**	Previous NRC Exam Other Exam
Reference(s):	ON-119	
Learning Objective:	PLOT-5036 3a	
K/A System:	300000 Instrument Air	Importance; RO 2.8
K/A Statement:	K4.01 - Knowledge of Instrument Air design feature(s) and/or interlocks which provide for the following: Manual/automatic transfers of control	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

9 **ID: 1649631** **Points: 1.00**

During performance of GP-11.E, Reactor Protection System - Scram and ARI Reset, the URO takes the Scram Reset switch to the Group 1 and 4 position.

Which one of the following describes the response of the white scram group lights on Panel C015, "System "A" PCIS/RPS" panel and C017, "System "B" PCIS/RPS" panel?

A. All four (4) lights on C015 illuminate, only.

B. All four (4) lights on C017 illuminate, only.

C. Two (2) lights on C015 illuminate and two (2) lights on C017 illuminate only.

D. NO lights illuminate until the Scram Reset switch is also taken to the Group 2 and 3 position.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	When the Scram Reset switch is taken to the position 1 and 4, it closes the scram inlet and outlet valves for two of the four rod groups (resets half scram). The white lights on both the C015 and C017 panels are indications of the rod group reset.
Distracters:	A	Two lights on each panel illuminate, not all four on C015. Plausible if candidate associates this switch position with resetting all RPS A logic.
	B	Two lights on each panel illuminate, not all four on C017. Plausible if candidate associates this switch position with resetting all RPS B logic.
	D	Two lights on each panel illuminate as soon as the switch is taken to either position. Plausible if the candidate believes the full scram logic must be reset to illuminate these lights. GP-11.E does not have the operator check lights until after the switch is taken to both positions, at which time all are illuminated.

Question 9 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649631
User-Defined ID:	
Cross Reference Number:	
Topic:	ILT-5060F-9g-002 2015 revised for 2017
Num Field 1:	2015 NRC
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	New Exam item **(2015 NRC #19)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	GP-11E, M-1-S-54	
Learning Objective:	PLOT-5060F 9g	
K/A System:	212000 RPS	Importance; RO 3.3
K/A Statement:	K5.02 - Knowledge of the operational implications of the following concepts as they apply to RPS: Specific logic arrangements	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

10 **ID: 1682360** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- A loss of ALL offsite power occurs.
- Concurrently, a steam leak in the Drywell results in a Drywell pressure of 3 psig and rising rapidly.
- Reactor pressure is 400 psig and down slow.
- All Emergency Diesel Generators (EDGs) automatically start.
- All EDG output breakers close simultaneously.

Which one of the following describes the load starting sequence once the EDGs restore power?

A. Core Spray pumps start, then ESW pumps start, then RHR pumps start

B. RHR pumps start, then ESW pumps start, then Core Spray pumps start

C. RHR pumps start, then Core Spray pumps start, then ESW pumps start

D. Core Spray pumps start, then RHR pumps start, then ESW pumps start

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	With a combined LOOP / LOCA, EDGs re-energize buses and then ECCS equipment loads sequentially to prevent overloading the EDGs. RHR pumps start first. Next, Core Spray pumps start. Finally, ESW pumps start.
Distracters:	A	Core Spray pumps start after RHR pumps. Plausible if the candidate confuses the order and since both pumps are needed for injection.
	B	ESW pumps start after RHR pumps. Plausible if the candidate confuses the order and since ESW provides cooling for both ECCS coolers and the EDGs.
	D	ESW pumps start after Core Spray pumps. Plausible if the candidate confuses the order and since ESW provides cooling for both ECCS coolers and the EDGs.

Question 10 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1682360
User-Defined ID:	
Cross Reference Number:	264000 K5.06
Topic:	5054 7b EDG loadingi
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(8)

Source Documentation

Source:	New Exam item Modified Bank **(SSES LOC27 NRC #10)** Exam Bank ILT Exam Bank	Previous NRC Exam Other
Reference(s):	SO 54.7.E	
Learning Objective:	PLOT-5054 7b	
K/A System:	264000 EDGs	Importance; RO 3.4
K/A Statement:	K5.06 - Knowledge of the operational implications of the following concepts as they apply to EDGs: Load sequencing	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

11 **ID: 1588379** **Points: 1.00**

Unit 2 has scrammed with the following:

- RCIC is operating for level control.
- A steam leak from RCIC causes an automatic RCIC Isolation on high area temperature.
- The cause of the leak is corrected and the area is being ventilated.
- You are directed to restore RCIC from the isolation in accordance with SO 13.7.A-2, Recovery from RCIC System Isolation or Turbine Trip.
- Reactor water level is -50" and slowly lowering.

Which one of the following describes the response of RCIC once the high area temperature condition clears, in accordance with SO 13.7.A-2?

Once the high area temperature condition clears, the isolation signal...

A. automatically resets and MO-15 and MO-16, Steam Isol Valves, automatically re-open.

B. automatically resets, but MO-15 and MO-16, Steam Isol Valves, must be manually re-opened.

C. remains sealed-in until manually reset. Then, MO-15 and MO-16, Steam Isol Valves, automatically re-open.

D. remains sealed-in until manually reset. Then, MO-15 and MO-16, Steam Isol Valves, must be manually re-opened.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	MO-15 and MO-16 remain closed until the isolation signal is manually reset by depressing the Div I and II Auto Isolation Reset pushbuttons. Since an initiation signal is present (level <-48"), MO-15 and MO-16 then automatically open. **Note:** The question meets the K/A by testing knowledge of a RCIC design feature that prevents release of radioactive steam to the Reactor Building (high area temperature isolation) and how this design feature operates when the initiating condition clears to allow the system to go back into operation.
Distracters:	A	The isolation signal remains sealed-in until manually reset. Plausible because the high temperature condition will clear with ventilation in progress, but the logic has a seal-in feature.
	B	The isolation signal remains sealed-in until manually reset. Plausible because the high temperature condition will clear with ventilation in progress, but the logic has a seal-in feature.
	D	Since an initiation signal is present (level <-48"), MO-15 and MO-16 then automatically open. Plausible because if an initiation signal was not present, then these valves would remain closed until manually opened.

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588379
User-Defined ID:	
Cross Reference Number:	
Topic:	RCIC high temp isolation reset with initiation signal
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation

Source:	X New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	SO 13.7.A-2, SO 13.1.C-2	
Learning Objective:	PLOT-5013 3e	
K/A System:	217000 RCIC	Importance; RO 3.2
K/A Statement:	K4.05 - Knowledge of RCIC design feature(s) and/or interlocks which provide for the following: Prevents radioactivity release to auxiliary/reactor building	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

12 **ID: 1588388** **Points: 1.00**

Unit 2 has experienced a failure to scram with the following:

- A loss of all offsite power has occurred.
- The E-1 and E-2 diesel generators have failed to start.

Which one of the following describes the resulting availability of Standby Liquid Control pumps 2A and 2B to inject boron into the Reactor?

A. Both pumps are available to inject.

B. 2A is available to inject, but 2B is NOT.

C. 2B is available to inject, but 2A is NOT.

D. NEITHER 2A NOR 2B are available to inject.

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	The LOOP and failure of EDGs E-1 and E-2 to start result in 4KV buses E12 and E22 being de-energized. This results in load centers E124 and E224 being de-energized. SLC pump 2A is powered from E124 and SLC pump 2B is powered from E224. Therefore, neither pump is available to inject.
Distracters:	A	The given AC power losses result in neither pump being available to inject. Plausible because two other emergency load centers still have power (E324 and 424).
	B	Pump 2B is not available to inject due to loss of E224 from E-2 failure to start. Plausible because two other emergency load centers still have power (E324 and 424).
	C	Pump 2A is not available to inject due to loss of E124 from E-1 failure to start. Plausible because two other emergency load centers still have power (E324 and 424).

Question 12 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588388
User-Defined ID:	
Cross Reference Number:	
Topic:	LOOP and EDG A/B failure
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	COL 11.1.A-2, E-1 sheet 1	
Learning Objective:	PLOT-5011 7c	
K/A System:	211000 SLC	Importance; RO 3.2
K/A Statement:	K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the SLC: A.C. power	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

13 **ID: 1658969** **Points: 1.00**

Unit 2 is operating at 100% with a Traversing In-Core Probe (TIP) detector in the core.

Which one of the following describes the status of the ball valve and the purge valve following a Group II isolation?

	Ball Valve	Purge Valve
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	If a PCIS Group II isolation signal is received while any TIP detectors are outside of their shield, the detector(s) will withdraw to the "in-shield" position and the associated ball valve will close. The isolation signal also closes the TIP purge valve.
Distracters:	A	The purge valve and the Ball valve both close. Plausible if the candidate does not recall the logic or the purpose of the Purge valve and Ball valve.
	B	The ball valve also closes. Plausible if the candidate does not recall the logic or the purpose of the Ball valve.
	C	The Purge valve also closes. Plausible if the candidate does not recall the logic or the purpose of the Purge valve.

Question 13 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1658969
User-Defined ID:	
Cross Reference Number:	223002 A1.02
Topic:	ILT-5007F 3a-005
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(9)

Source Documentation

Source:	New Exam item **X** Modified Bank **1588389** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	GP-8.B COL, SO 7F.7.A	
Learning Objective:	PLOT-5007F 3a	
K/A System:	223002 PCIS/Nuclear Steam Supply Shutoff	Importance; RO 3.7
K/A Statement:	A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the PCIS/Nuclear Steam Supply Shutoff controls including: Valve closures	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

14 **ID: 1588390** **Points: 1.00**

A Unit 2 shutdown is in progress with the following:

- Preparations are underway to place RHR in the Shutdown Cooling (SDC) mode.
- Reactor pressure is 500 psig and slowly lowering on Turbine Bypass Valves.

Which one of the following identifies the ***approximate*** maximum Reactor pressure at which SDC flow can be established to the Reactor?

A. 450 psig

B. 300 psig

C. 100 psig

D. 45 psig

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is approximately 45 psig due to interlocks on the SDC IVs. **Note:** The K/A requires predicting and/or monitoring changes in RHR pump discharge pressure or system discharge pressure associated with operating SDC controls. This is accomplished by testing the ability to predict/monitor the RHR system discharge pressure (which equates to Reactor pressure while in SDC) required to allow successful operation SDC controls (initiating flow by manipulating RHR pumps/valves).
Distracters:	A	The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is approximately 45 psig due to interlocks on the SDC IVs. 450 psig is based on the highest Reactor pressure at which the LPCI injection valves will open in the LPCI mode of RHR.
	B	The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is approximately 45 psig due to interlocks on the SDC IVs. 300 psig is based on the highest approximate Reactor pressure at which RHR pumps are capable of injecting to the Reactor.
	C	The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is approximately 45 psig due to interlocks on the SDC IVs. 100 psig is based on the highest Reactor pressure at which SO 10.1.B-2 allows filling the SDC suction piping.

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588390
User-Defined ID:	
Cross Reference Number:	
Topic:	Highest Reactor pressure to place SDC in service
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(3)

Source Documentation

Source:	New Exam item **(SSES LOC27 NRC #14)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	SO 10.1.B-2, ARC-224 C-1	
Learning Objective:	PLOT-5010 3n	
K/A System:	205000 Shutdown Cooling	Importance; RO 2.8
K/A Statement:	A1.09 - Ability to predict and/or monitor changes in parameters associated with operating the Shutdown Cooling controls including: SDC/RHR pump/system discharge pressure	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

15 **ID: 1588391** **Points: 1.00**

A steam leak in the Unit 3 Reactor Building results in the following:

- Standby Gas Treatment (SBGT) automatically initiates.
- Annunciator ARC-317 L-2, STDBY GAS FILTER HEATER FAILURE, alarms.
- Investigation reveals that SBGT Heater 0AE65 has failed.

Which one of the following describes (1) the concern with this heater failure and (2) the extent of the impact of this failure on SBGT?

A. (1) Charcoal filter efficiency may be reduced.
(2) Only one train is affected.

B. (1) Charcoal filter efficiency may be reduced.
(2) Both trains are affected.

C. (1) Fan performance may be degraded.
(2) Only one train is affected.

D. (1) Fan performance may be degraded.
(2) Both trains are affected.

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	Failure of this heater will result in higher than normal filter train moisture content. High filter train moisture content is a concern because it degrades charcoal filter efficiency. This heater failure affects only the A train of SBGT (the B train has a separate heater). **Note:** The question meets the K/A by testing ability to predict the effect of high train moisture content on SGTS (loss of heater results in high moisture content in process stream, which degrades filter performance). There is very limited procedural guidance available related to high filter train moisture content. The 2nd half of the K/A is addressed as well as possible by testing knowledge of the extent of the given failure, which is necessary to mitigate the failure (would be useful in determining which train to remove from service).
Distracters:	B	Only one train is affected. The given heater is located in the A filter assembly and a separate heater is located in the B filter assembly. Plausible that this is a shared heater for both filter assemblies.
	C	The concern is charcoal filter efficiency, not fan performance. Plausible that the heater is necessary for fan performance (either for moisture removal to prevent blade impingement or for proper temperature conditions to be maintained).
	D	The concern is charcoal filter efficiency, not fan performance. Plausible that the heater is necessary for fan performance (either for moisture removal to prevent blade impingement or for proper temperature conditions to be maintained). Only one train is affected. The given heater is located in the A filter assembly and a separate heater is located in the B filter assembly. Plausible that this is a shared heater for both filter assemblies.

Question 15 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588391
User-Defined ID:	
Cross Reference Number:	
Topic:	Filter heater failure effect
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(9)

Source Documentation

Source:	New Exam item Modified Bank Bank **(NMP1 2013 Cert #46)** ILT Exam Bank	Previous NRC Exam **X** Other Exam
Reference(s):	DBD P-S-32, M-397	
Learning Objective:	PLOT-5009A 10c	
K/A System:	261000 SBGT	Importance; RO 2.5
K/A Statement:	A2.04 - Ability to (a) predict the impacts of the following on the SGTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train moisture content	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

16 **ID: 1650226** **Points: 1.00**

Both Units are operating at 100% power with the following:

- The B ESW pump breaker is racked out for corrective maintenance.
- A loss of all offsite power occurs.

Which one of the following describes the resulting response of the Emergency Diesel Generators (EDGs)?

A. Only two EDGs start.

B. All four EDGs start and are supplied with cooling water from ESW.

C. All four EDGs start. Two are supplied with cooling water from ESW. Two are supplied with cooling water from ECW.

D. All four EDGs start. Two are supplied with cooling water from ESW. Two are running without cooling water.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	All four EDGs start and are supplied with cooling water from the A ESW pump. The ESW supply to the EDGs is not divisionalized. Rather, both the A and B ESW pumps discharge into a common header to all four EDGs.
Distracters:	A	All four EDGs start. Plausible that ESW pumps would be divisionalized such that two EDGs lost ESW and would not start without adequate cooling water pressure.
	C	All four EDGs are supplied with cooling water from ESW. Plausible that ESW pumps would be divisionalized such that two EDGs lost ESW and are automatically supplied from ECW, which is an automatic backup if all ESW is lost.
	D	All four EDGs are supplied with cooling water from ESW. Plausible that ESW pumps would be divisionalized such that two EDGs lost ESW and alternate cooling would need to be manually aligned or not available at all.

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650226
User-Defined ID:	
Cross Reference Number:	
Topic:	B ESW pump loss effect on EDGs
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(8)

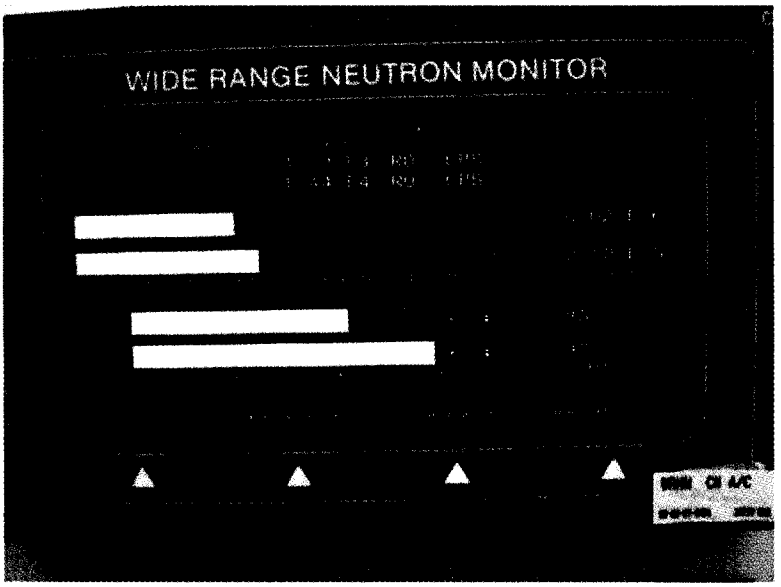
Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	M-315, SO 54.7.E	
Learning Objective:	PLOT-5033 6a	
K/A System:	400000 Component Cooling Water	Importance; RO / SRO 2.9
K/A Statement:	K3.01 - Knowledge of the effect that a loss or malfunction of the Component Cooling Water will have on following: Loads cooled by CCWS	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

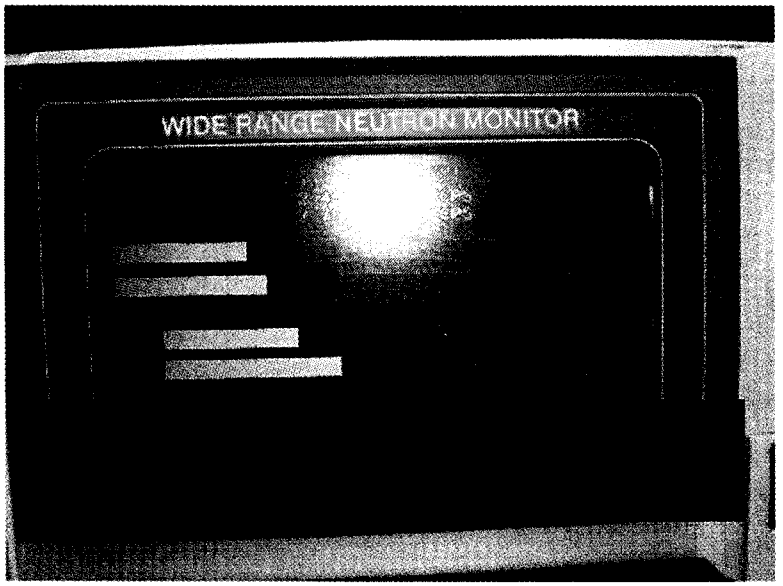
17 **ID: 1588454** **Points: 1.00**

A Unit 2 startup is in progress with the following WRNM indications:

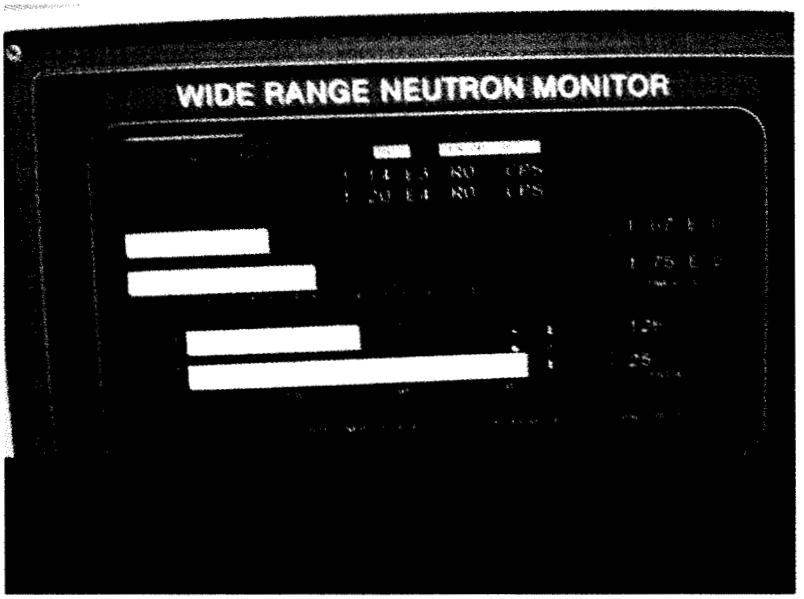
WRNM Channel A/C



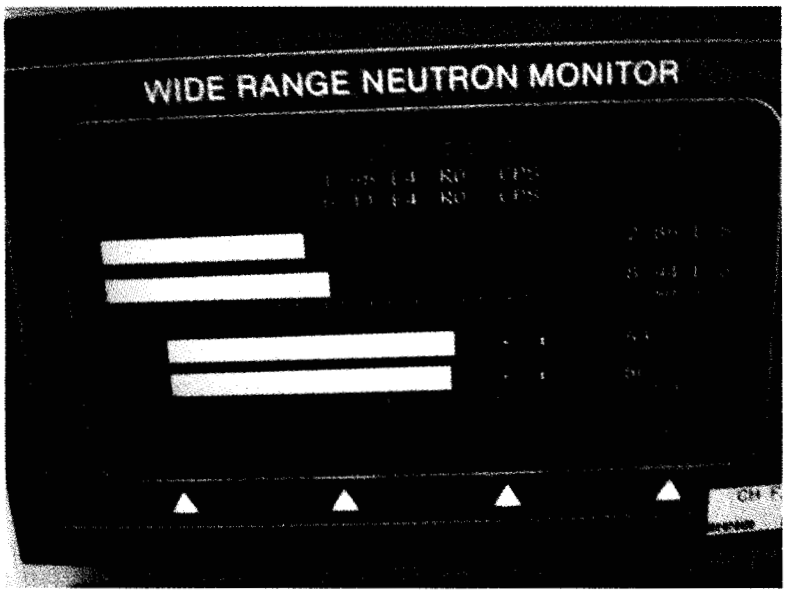
WRNM Channel B/D



WRNM Channel E/G



WRNM Channel F/H



Which one of the following describes the implication of these indications?

A. All WRNM indications are within the administrative limit of GP-2, Normal Plant Startup.

B. One or more WRNM indications have exceeded an administrative limit of GP-2, Normal Plant Startup. NO control rod block is received based on these indications.

C. A control rod block is received based on these indications, but NO half scram signals are generated.

D. A half scram is signal is received based on these indications.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	A control rod block is received because one WRNM is indicating a Reactor period of <28 seconds. No half scram signals are received because no WRNM is indicating a Reactor period of <19 seconds.
Distracters:	A	GP-2 requires maintaining Reactor period on all WRNMs >50 seconds. Multiple WRNMs are indicating <50 seconds. Plausible because some of the WRNMs are indicating >50 seconds and other WRNM limits are not being exceeded (19 second scram).
	B	A control rod block is received because one WRNM is indicating a Reactor period of <28 seconds. Plausible because no other WRNM is <28 seconds and other WRNM limits are not being exceeded (19 second scram).
	D	No half scram signals are received because no WRNM is indicating a Reactor period of <19 seconds. Plausible because Reactor period is high and causing a rod block.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588454
User-Defined ID:	
Cross Reference Number:	
Topic:	WRNM indications - rod block, no scram
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation		
Source:	X New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	GP-2, ARC-210 F-3 and G-3	
Learning Objective:	PLOT-5060C 9j	
K/A System:	215003 IRM	Importance; RO 3.3
K/A Statement:	A3.01 - Ability to monitor automatic operations of the IRM including: Meters and recorders	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

18 **ID: 1659121** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

Time (hh:mm)	Condition
00:00	A steam leak develops in the Drywell. Drywell pressure is 2.1 psig and slowly rising.
01:15	A loss of all offsite power occurs.
02:30	All offsite power sources re-energize.

Which one of the following describes when 4KV buses E12, E22, E32, and E42 first transfer to the Emergency Diesel Generators (EDGs) and the response of these buses at time 02:30?

(1) 4KV buses E12, E22, E32, and E42 first transfer to the EDGs at time...
(2) At time 02:30, these buses...

A. (1) 00:00
(2) remain energized by the EDGs.

B. (1) 00:00
(2) automatically transfer back to offsite power.

C. (1) 01:15
(2) remain energized by the EDGs.

D. (1) 01:15
(2) automatically transfer back to offsite power.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	At time 00:00, a LOCA signal is received due to Drywell pressure above 2 psig. This causes the EDGs to start, but their output breakers do not close in on the 4KV buses because they are still energized by offsite power. At time 01:15, the 4KV buses de-energize due to the loss of offsite power and are re-energized automatically by the EDGs. When offsite power is restored at time 02:30, the 4KV buses remain powered from the EDGs. Manual action is required to transfer these buses back to their normal power source.
Distracters:	A	The 4KV buses first transfer to the EDGs at time 01:15. Plausible because the EDGs first start at time 00:00, but do not load onto the 4KV buses since no undervoltage signal exists.
	B	The 4KV buses first transfer to the EDGs at time 01:15. Plausible because the EDGs first start at time 00:00, but do not load onto the 4KV buses since no undervoltage signal exists. When offsite power is restored at time 02:30, the 4KV buses remain powered from the EDGs. Manual action is required to transfer these buses back to their normal power source. Plausible because the swap from offsite power to the EDGs is automatic and some electrical swap features are normal-seeking such that they would automatically swap back.
	D	When offsite power is restored at time 02:30, the 4KV buses remain powered from the EDGs. Manual action is required to transfer these buses back to their normal power source. Plausible because the swap from offsite power to the EDGs is automatic and some electrical swap features are normal-seeking such that they would automatically swap back.

Question 18 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659121
User-Defined ID:	
Cross Reference Number:	
Topic:	LOCA, then LOOP, then power restoration
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	SE-11, SO 52B.2.B, SO 54.7.E	
Learning Objective:	PLOT-5054 5a	
K/A System:	262001 AC Electrical Distribution	Importance; RO 3.2
K/A Statement:	A3.02 - Ability to monitor automatic operations of the AC Electrical Distribution including: Automatic bus transfer	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

19 **ID: 1140325** **Points: 1.00**

The 2B Reactor Feed Pump (RFP) is being started per SO 6C.1.C-2, Startup of Second or Third Reactor Feedwater Pump.

The following indications exist for the 2B RFP:

- Speed is 2800 RPM.
- MSC SELECT is lit.
- M/A PERMISSIVE is lit.
- M/A SELECT is NOT lit.
- M/A is in MANUAL.

Which one of the following describes the status of the 2B RFP?

The 2B RFP is ready to be transferred to ___(1)___.
In order to complete the transfer, the operator must depress ___(2)___.

A. (1) the M/A Station
(2) M/A SELECT

B. (1) the M/A Station
(2) AUTO on the M/A Station

C. (1) the Master Level Controller
(2) AUTO on the M/A Station

D. (1) the Master Level Controller
(2) AUTO on the Master Level Controller

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	Per SO 6C.1.C-2, these are the indications expected prior to transferring RFP control from MSC to the M/A Station. The transfer is completed by depressing M/A SELECT.
Distracters:	B	Depressing AUTO on the M/A Station transfers RFP control to the Master Level Controller. Plausible if the candidate does not understand the actions required to transfer control to the M/A station. Depressing is a correct action to transfer M/A station control to the Master controller.
	C	RFP control must be transferred to the M/A Station before transferring to the MLC. M/A SELECT is lit and MSC SELECT is not lit when the M/A Station has control of the RFP. Plausible if the candidate does not understand the sequence required to transfer control to the MLC and believe that control can occur directly from the MSC.
	D	RFP control must be transferred to the M/A Station before transferring to the MLC. M/A SELECT is lit and MSC SELECT is not lit when the M/A Station has control of the RFP. Plausible if the candidate does not understand the sequence of step required in the transfer of the Feedwater system to master auto control.

Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1140325
User-Defined ID:	
Cross Reference Number:	
Topic:	ILT-5006-9g-001
Num Field 1:	2015 NRC
Num Field 2:	A NRC
Text Field:	2009 NRC exam question 20

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item **(2015 NRC #37)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	SO 6C.1.C-2	
Learning Objective:	PLOT-5006 9g	
K/A System:	259002 Reactor Water Level Control	Importance; RO 2.8
K/A Statement:	A4.01 - Ability to manually operate and/or monitor in the control room: All individual component controllers in the manual mode	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

20 **ID: 1659000** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Battery Charger 2AD003 is placed in the equalize charge mode in accordance with SO 57B.1-2, 125/250 Volt Station Battery Charger Operations.
- During the charge, AC power to the charger is lost due to a momentary loss of power to the E-12 bus.
- Power is subsequently restored to the E-12 bus by the Emergency Diesel Generator.

The 2A Battery Charger output breaker will be __(1)__ and will be in the __(2)__ mode.

A. (1) open
(2) float

B. (1) open
(2) equalizing

C. (1) closed
(2) float

D. (1) closed
(2) equalizing

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	From Note 2 in SO 57B.1-2: "Upon a loss of AC input power, the battery charger will return to the same mode it was in once power is restored. IF the battery charger was in the Equalize mode, THEN the timer will pick up where it was interrupted AND time out." The output breaker doesn't get a trip signal on a loss of power.
Distracters:	A	The output breaker doesn't get a trip signal on a loss of power. Plausible if the candidate does not remember the breaker logic for the battery charger. The charger will return to the equalize charge mode. Plausible if the applicant remembers the charger will automatically restart but does not remember it will return to the same mode it was in prior to the power loss.
	B	The output breaker doesn't get a trip signal on a loss of power. Plausible if the candidate does not remember the breaker logic for the battery charger.
	C	The charger will return to the equalize charge mode. Plausible if the applicant remembers the charger will automatically restart but does not remember it will return to the same mode it was in prior to the power loss.

Question 20 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	1.00
System ID:	1659000
User-Defined ID:	ILT-5057-7A-004
Cross Reference Number:	263000 K1.01
Topic:	ILT-5057-7A-005
Num Field 1:	
Num Field 2:	A NRC
Text Field:	NRC-09-1

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item **(2011 NRC #13)** **X** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	SO 57B.1-2	
Learning Objective:	PLOT-5057 7a	
K/A System:	263000 DC Electrical Distribution	Importance; RO 3.3
K/A Statement:	K1.01 - Knowledge of the physical connections and/or cause-effect relationships between DC Electrical Distribution and the following: A.C. electrical distribution	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

21 **ID: 1659007** **Points: 1.00**

Unit 2 has experienced a scram from 10% power.

The URO makes the following scram reports:

- Reactor Mode switch is in Shutdown
- All control rods are inserting
- APRMs are down scale
- RPV level is 20 inches and rising slowly, the "C" RFP is available
- RPV level reached a lowest value of 5 inches during the transient
- RPV pressure is 900 psig and EHC is available

Following the report, the "D" SRV fails open.

Based on this information, the Crew __(1)__ enter and execute OT-114, "Inadvertent Opening of a Relief Valve" and __(2)__ enter and execute T-101, "RPV Control".

A. (1) shall
(2) shall

B. (1) shall
(2) shall NOT

C. (1) shall NOT
(2) shall

D. (1) shall NOT
(2) shall NOT

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	An entry condition exists for OT-114. OT-114 is not one of the OT procedures that requires exit on a scram or when another OT is entered. The scram occurred from low power. RPV level is not expected to drop below 1" (the T-101 entry condition) and the URO's report confirms that expectation by reporting that RPV level is 20 inches and rising without a RFP in service.
Distracters:	A	An entry condition exists for OT-114. OT-114 is not one of the OT procedures that requires exit on a scram or when another OT is entered. An entry condition for T-101 does not exist because RPV level did not go below 1 inch. Plausible if the candidate does not recall the T-101 entry conditions.
	C	An entry condition exists for OT-114. OT-114 is not one of the OT procedures that requires exit on a scram or when another OT is entered. Plausible if the candidate does not recall that OT-114 is an OT that is executed even after a reactor scram. An entry condition for T-101 does not exist because RPV level did not go below 1 inch. Plausible if the candidate does not recall the T-101 entry conditions.
	C	An entry condition exists for OT-114. OT-114 is not one of the OT procedures that requires exit on a scram or when another OT is entered. Plausible if the candidate does not recall that OT-114 is an OT that is executed even after a reactor scram.

Question 21 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	1.00
System ID:	1659007
User-Defined ID:	
Cross Reference Number:	
Topic:	ILT 1540-1-007 OT-114/T-101
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (10)

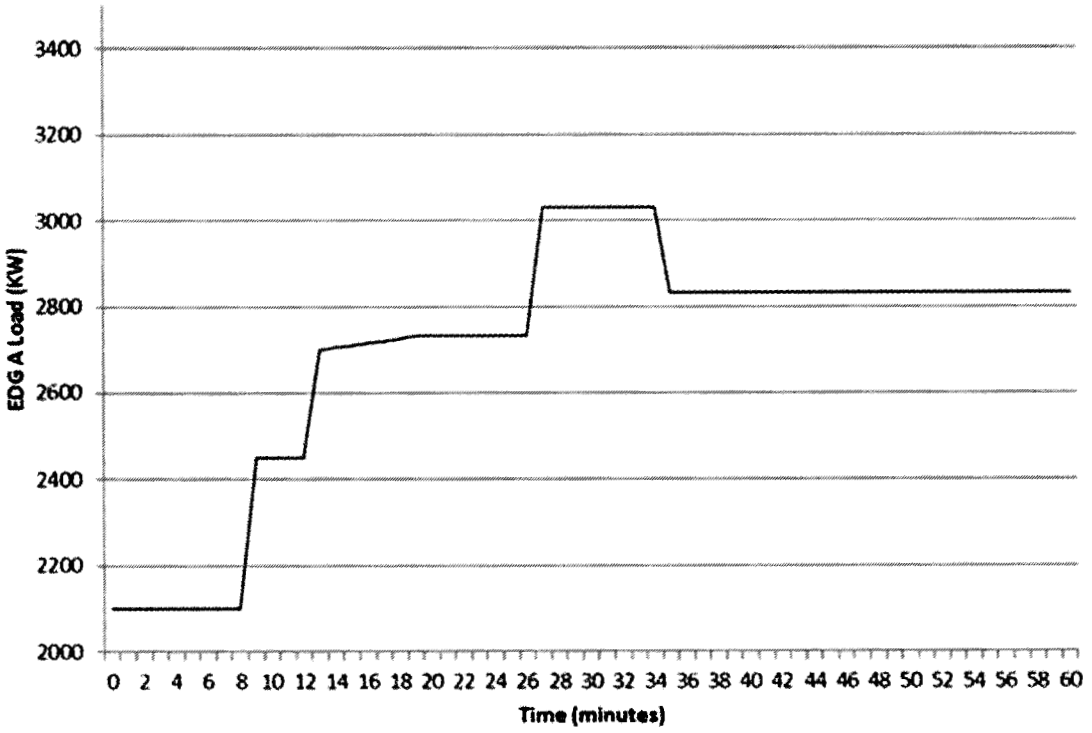
Source Documentation

Source:	New Exam item Previous NRC Exam **X** Modified Bank **1649748** Other Exam Bank ILT Exam Bank	
Reference(s):	OT-114, T-101, T-102	
Learning Objective:	PLOT-DBIG-1540 1	
K/A System:	218000 ADS	Importance; RO 3.8
K/A Statement:	2.4.8 - Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

22 **ID: 1588478** **Points: 1.00**

Unit 2 has experienced a loss of all offsite power with the following:

- Only the E-1 EDG has started.
- The E-1 EDG loading is shown in the following graph:



Which one of the following describes the loading of the E-1 EDG during this time period, in accordance with SO 52A.1.B, Diesel Generator Operations?

The E-1 EDG loading __(1)__ been below the 30 minute load rating for the entire time period.
The E-1 EDG loading is currently __(2)__ the continuous load rating.

A. (1) has
(2) below

B. (1) has
(2) above

C. (1) has NOT

(2) below

D. (1) has NOT
(2) above

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	SO 52A.1.B give a continuous load rating of 2600 KW and a 30 minute load rating of 3100-3250 KW. The E-1 EDG loading has been below the 30 minute load rating of 3100-3250 KW during the entire time period (~3050 KW peak). The E-1 EDG loading is currently above the continuous load rating of 2600 KW. **Note:** The question meets the K/A by providing a trend of EDG loading and requiring the candidate to assess the trend-line against the appropriate control room reference material (memory level knowledge of SO 52A.1.B requirements for EDG loading).
Distracters:	A	The E-1 EDG loading is currently above the continuous load rating of 2600 KW. Plausible because loading is below all other limits, including the 2600-3000 KW limit for 2000 hours.
	C	The E-1 EDG loading is currently above the continuous load rating of 2600-3000 KW. Plausible because loading is below all other limits, including the 3000 KW limit for 2000 hours.
	D	The E-1 EDG loading has been below the 30 minute load rating of 3100-3250 KW during the entire time period (~3050 KW peak). Plausible because loading did go above multiple limits, including the 2600-3000 KW limit for 2000 hours.

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588478
User-Defined ID:	
Cross Reference Number:	
Topic:	EDG load limits
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(8)

Source Documentation

Source:	X New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	SO 52A.1.B	
Learning Objective:	PLOT-5052 9j	
K/A System:	264000 EDGs	Importance; RO 4.2
K/A Statement:	2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

23 **ID: 1650228** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- A Reactor Building Closed Cooling Water (RBCCW) malfunction is causing high system temperatures.
- ON-113, Loss of RBCCW, has been entered.

Which one of the following identifies a component bearing temperature that must be monitored and the required action if this component temperature exceeds the limit in accordance with ON-113?

Monitor __(1)__ bearing temperatures.
If temperatures exceed the limit in ON-113, then __(2)__.

A. (1) Recirc pump motor
(2) trip the affected pump

B. (1) Recirc pump motor
(2) reduce pump speed to control temperature

C. (1) Feedwater pump/turbine
(2) trip the affected pump/turbine

D. (1) Feedwater pump/turbine
(2) reduce pump/turbine speed control temperature

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	High RBCCW temperature challenges cooling of Recirc pump motor bearings. ON-113 requires monitoring these temperatures. If they exceed 194°F, then the affected pump must be tripped.
Distracters:	B	The pump must be tripped. Plausible because Recirc pump seal cavity temperatures also must be monitored per ON-113 and if they exceed limits, then pump speed is lowered to control temperatures.
	C	Feedwater pump/turbine bearing temperatures are not monitored for high RBCCW temperature. Plausible because they would be affected by high temperatures in Service Water, which is related to RBCCW, and also are monitored in the Control Room.
	D	Feedwater pump/turbine bearing temperatures are not monitored for high RBCCW temperature. Plausible because they would be affected by high temperatures in Service Water, which is related to RBCCW, and also are monitored in the Control Room. The pump must be tripped. Plausible because Recirc pump seal cavity temperatures also must be monitored per ON-113 and if they exceed limits, then pump speed is lowered to control temperatures.

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650228
User-Defined ID:	
Cross Reference Number:	
Topic:	ON-113 - Component to monitor and action for high temp
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:				
	Psychometrics			
	Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
	MEMORY			10CFR55.41(b)(4)
	Source Documentation			
	Source:	**X** New Exam item Modified Bank ILT Exam Bank		Previous NRC Exam Other Exam Bank
	Reference(s):	ON-113		
	Learning Objective:	PLOT-5035 10c		
	K/A System:	400000 Component Cooling Water		Importance; RO 2.9
	K/A Statement:	A2.03 - Ability to (a) predict the impacts of the following on the Component Cooling Water; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High/low CCW temperature		
	REQUIRED MATERIALS:	**NONE**		
	Notes and Comments:			

24 **ID: 1588480** **Points: 1.00**

Unit 2 is operating at 100% power when the 2A 125 VDC power distribution panel 20D021 de-energizes due to a sustained electrical fault.

Based on this information, the 4 KV breaker associated with the E-12 bus __(1)__ be operated from the control room and automatic breaker trip capability __(2)__ maintained.

A. (1) can
(2) is

B. (1) can
(2) is NOT

C. (1) can NOT
(2) is

D. (1) can NOT
(2) is NOT

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	2A 125 VDC power distribution panel 20D021 provides all DC control power for Bus E-12 breakers. Upon loss of this DC power, remote control of the breaker from the Control Room and all automatic protective functions are lost.
Distracters:	A	Breakers cannot be operated from the control room. Plausible if the candidate doesn't recall the function of DC control power. Breakers will not automatically trip with out control power. Plausible if the candidate doesn't recall the function of DC control power or recall the function of 4 KV breakers
	B	Breakers cannot be operated from the control room. Plausible if the candidate doesn't recall the function of DC control power.
	C	Breakers will not automatically trip with out control power. Plausible if the candidate doesn't recall the function of DC control power or recall the function of 4 KV breakers

Question 24 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588480
User-Defined ID:	
Cross Reference Number:	
Topic:	Effect on E-12 breakers of control power loss
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item Previous NRC Exam Modified Bank Other Exam Bank **X** ILT Exam Bank	
Reference(s):	SE-13 Attachment 2	
Learning Objective:	PLOT-5054 7a	
K/A System:	263000 DC Electrical Distribution	Importance; RO 3.3
K/A Statement:	A4.01 - Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

25 **ID: 1588532** **Points: 1.00**

Unit 2 has experienced a loss of coolant accident with the following:

- RCIC is injecting 400 gpm to the Reactor.
- CST level is 5' and slowly lowering.
- Torus water level is 10.8' and slowly lowering.

Which one of the following describes the effect of these conditions on the operation of RCIC?

RCIC is currently operating with suction from the...

A. CST. If CST level continues to lower, the RCIC suction path will automatically swap to the Torus.

B. Torus. If Torus level continues to lower, the RCIC suction path will automatically swap to the CST.

C. CST. If CST level continues to lower, the RCIC suction path will stay aligned to the CST and RCIC pump vortexing may become a concern.

D. Torus. If Torus level continues to lower, the RCIC suction path will stay aligned to the Torus and RCIC pump vortexing may become a concern.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	RCIC suction is normally aligned to the CSTs. However, RCIC suction has already swapped to the Torus due to CST level < 5.25'. Torus water level is below the EOP entry and Tech Spec low level and approaching the 10.5' action level in T-102, but there is no automatic swap back to the CSTs once RCIC suction has swapped to the Torus. RCIC pump vortexing becomes a concern if Torus water level continues to lower.
Distracters:	A	RCIC suction is currently from the Torus because CST level is <5.25'. Plausible because the CST is the normal suction source and Torus level is also low.
	B	There is no auto swap from the CSTs to the Torus. Plausible because there is an auto swap in the other direction and low Torus water level also is a concern for system operation.
	C	RCIC suction is currently from the Torus because CST level is <5.25'. Plausible because the CST is the normal suction source and Torus level is also low.

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588532
User-Defined ID:	
Cross Reference Number:	
Topic:	RCIC suction with low level in both CST and Torus
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(8)

Source Documentation

Source:	New Exam item **(JAF 9/14 NRC #12)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	DBD P-S-39, T-102 Bases	
Learning Objective:	PLOT-5013 7c	
K/A System:	217000 RCIC	Importance; RO 3.5
K/A Statement:	K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the RCIC: Suppression pool water supply	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

26 **ID: 1600733** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- A momentary loss of output from the Static Inverter to Panel 20Y050 causes the static switch to bypass the inverter.
- Annunciator 220 F-5, INVERTER TROUBLE, alarms.

Then, normal power output from the Static Inverter is restored.

Which one of the following describes effect of this transient on Panel 20Y050 and the corresponding Control Room indication?

Panel 20Y050...

A. must be manually transferred back to the Static Inverter. The Control Room alarm automatically resets when the transfer occurs.

B. must be manually transferred back to the Static Inverter. The Control Room alarm does NOT reset until the local alarm is manually reset.

C. automatically transfers back to the Static Inverter. The Control Room alarm automatically resets when the transfer occurs.

D. automatically transfers back to the Static Inverter. The Control Room alarm does NOT reset until the local alarm is manually reset.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	The static switch is normal seeking. It will automatically transfer power back to the Static Inverter 30 seconds after the Static Inverter output is restored. The Control Room alarm will then automatically reset.
Distracters:	A	The static switch is normal seeking. It will automatically transfer power back to the Static Inverter 30 seconds after the Static Inverter output is restored. Plausible that the design of the static switch would require manual action to return to the initially degraded power source to prevent damage.
	B	The static switch is normal seeking. It will automatically transfer power back to the Static Inverter 30 seconds after the Static Inverter output is restored. Plausible that the design of the static switch would require manual action to return to the initially degraded power source to prevent damage. The Control Room alarm will automatically reset when the transfer occurs. Plausible because this requires local alarm reset on some plants.
	D	The Control Room alarm will automatically reset when the transfer occurs. Plausible because this requires local alarm reset on some plants.

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1600733
User-Defined ID:	
Cross Reference Number:	
Topic:	Return to normal power
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	New Exam item **(NMP1 2010 NRC #25)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	ARC-220 F-5, SO 58B.7.B-2	
Learning Objective:	PLOT-5058 7c	
K/A System:	262002 - UPS (AC/DC)	Importance; RO 2.8
K/A Statement:	A4.01 - Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

27 **ID: 1650249** **Points: 1.00**

During Refuel Floor operations on Unit 2, the Control Room receives the following alarm and indications:

- Annunciator 218 D-5, REAC BLDG OR REFUELING FLOOR VENT RAD MON DOWNSCALE, alarms.
- The following Refueling Floor Radiation Trip Units indicate downscale:
  - RIS-2-17-458 A
  - RIS-2-17-458 B

Which one of the following describes the impact of these downscale failures on Refuel Floor Ventilation?

Refuel Floor Ventilation…

A. isolates.

B. remains in service and CANNOT be isolated by the remaining Refueling Floor Radiation Trip Units.

C. remains in service and can be isolated by just one of the remaining Refueling Floor Radiation Trip Unit reaching the high setpoint.

D. remains in service and can be isolated by the remaining Refueling Floor Radiation Trip Units, but only if they both reach the high setpoint.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The Refueling Floor Radiation Trip Units isolate Refuel Floor Ventilation if a trip occurs on RIS-2-17-458 A or C accompanied by a trip on RIS-2-17-458 B or D. Downscale failure provides an alarm, but does not alter the logic. RIS-2-17-458 C and D both reaching the high setpoint will still isolate Refuel Floor Ventilation.
Distracters:	A	Downscale failure provides an alarm, but does not alter the logic. Plausible because downscale failure provides an alarm, reduces available trip units to cause the protective feature, and a fail-safe design feature could initiate the protective feature without harm for this system.
	B	RIS-2-17-458 C and D both reaching the high setpoint will still isolate Refuel Floor Ventilation. Plausible because if A and C or B and D failed downscale, this would be the correct answer.
	C	Downscale failure provides an alarm, but does not alter the logic. Plausible because downscale failure provides an alarm, reduces available trip units to cause the protective feature, and a fail-safe design feature could change the remaining logic to compensate for loss of trip units.

Question 27 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650249
User-Defined ID:	
Cross Reference Number:	
Topic:	Two Refuel Floor Rad monitors fail downscale
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Source Documentation

Level of Knowledge		RO
High		10CFR55.41(b)(7)
Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC-218 D-5 and D-4	
Learning Objective:	PLOT-5007G 4c	
K/A System:	288000 Plant Ventilation	Importance; RO 3.3
K/A Statement:	K1.05 - Knowledge of the physical connections and/or cause-effect relationships between Plant Ventilation and the following: Process radiation monitoring system	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

28 **ID: 1650231** **Points: 1.00**

A Unit 2 startup is in progress with the following:

- Reactor pressure is 350 psig.
- The 2A and 2C Condensate pumps are operating.
- The 2B Condensate pump is secured with its control switch green-flagged.
- NO Feedwater pumps are running yet.

Then, the 1 Aux Bus de-energizes due to a sustained electrical fault.

Which one of the following identifies the Condensate pumps that are operating one (1) minute later?

A. 2A only

B. 2C only

C. 2A and 2B

D. 2C and 2B

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	The 1 Aux Bus supplies power to the 2A Condensate pump. The 2B and 2C Condensate pumps are powered from the 2 Aux Bus. Therefore, the 2A Condensate pump trips and the 2C Condensate pump remains operating. The 2B Condensate pump remains in standby because there is no automatic start on trip of the 2A Condensate pump.
Distracters:	A	The 2A Condensate pump trips and the 2C Condensate pump remains operating. Plausible because this would be the correct answer for loss of the 2 Aux Bus.
	C	The 2A Condensate pump trips. Plausible if candidate mixed the power supplies for the 2A and 2C Condensate pumps. The 2B Condensate pump remains in standby. Plausible because many plants have auto-starts on Condensate pumps that would actuate in this situation and the 2B Condensate pump does have power.
	D	The 2B Condensate pump remains in standby. Plausible because many plants have auto-starts on Condensate pumps that would actuate in this situation and the 2B Condensate pump does have power.

Question 28 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650231
User-Defined ID:	
Cross Reference Number:	
Topic:	Loss of 1 Aux Bus
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(4)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC-203 E-2, F-2, and G2	
Learning Objective:	PLOT-5005 2a	
K/A System:	256000 Reactor Condensate	Importance; RO 2.7
K/A Statement:	K2.01 - Knowledge of electrical power supplies to the following: System pumps	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

29 **ID: 1670949** **Points: 1.00**

Unit 2 is shutdown with the following:

- RWCU is operating in Dump Mode to the Main Condenser.
- Dump flow is 100 gpm.
- Reactor water level is +20" and stable with Feedwater injecting in manual and AO 8091 in manual.

Then, a localized failure causes loss of Instrument Air pressure to just CV-55, RWCU Dump Flow.

In order to stabilize RPV level the operator will throttle _____.

A. closed AO 8091, "C Reactor Feedpump Bypass"

B. open AO 8091, "C Reactor Feedpump Bypass"

C. closed CV-4157, "RWCU Dump to Condenser"

D. open CV-4157, "RWCU Dump to Condenser"

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	On loss of Instrument Air, CV-55 fails closed. This lowers dump flow rate to 0 gpm. With Feedwater injecting in Manual, Reactor water level rises due to the loss of dump flow. The operator will need to throttle closed AO-8091
Distracters:	B	CV-55 fails closed, not open. Plausible because some air operated valves fail open on loss of air.
	C	CV-4157 is not controllable. It is a pressure control valve designed to automatically maintain downstream pressure. Plausible if the candidate does not understand the purpose of CV-4157 and believes based on the name that it can be throttled.
	D	CV-4157 is not controllable. It is a pressure control valve designed to automatically maintain downstream pressure. Plausible if the candidate does not understand the purpose of CV-4157 and believes based on the name that it can be throttled.

Question 29 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1670949
User-Defined ID:	
Cross Reference Number:	204000 K3.02
Topic:	Plot-5012 6b
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(3)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank Exam Bank ILT Exam Bank	Other
Reference(s):	M-354	
Learning Objective:	PLOT-5012 6b	
K/A System:	204000 RWCU	Importance; RO 3.1
K/A Statement:	K3.02 - Knowledge of the effect that a loss or malfunction of the RWCU will have on following: Reactor water level	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

30 **ID: 1601379** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- All Fuel Pool Service Water Booster pumps fail.
- Fuel Pool temperature is 89°F and rising slowly.
- Fuel Pool water level is normal.

Which one of the following identifies an alternate source of cooling water that can be supplied to the **Fuel Pool Cooling** heat exchangers?

A. ESW

B. RHR

C. HPSW

D. RBCCW

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	RBCCW may be manually aligned to provide an alternate source of cooling water to the FPC heat exchangers per AO 19.3-2.
Distracters:	A	ESW does not provide an alternate cooling source for the Fuel Pool Cooling heat exchangers. Plausible because it does provide a safety-related cooling water supply to the Emergency Diesel Generators and ECCS Room Coolers.
	B	RHR does not provide an alternate cooling source for the Fuel Pool Cooling heat exchangers. Plausible because RHR in the SDC mode can be used in some circumstances to augment FPC.
	C	HPSW does not provide an alternate cooling source for the Fuel Pool Cooling heat exchangers. Plausible because HPSW can be used to inject water into the Spent Fuel Pool as part of a feed and bleed operation.

Question 30 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601379
User-Defined ID:	
Cross Reference Number:	
Topic:	Backup to SW
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (13)

Source Documentation

Source:	New Exam item **(2008 NRC #35)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	AO 19.3-2	
Learning Objective:	PLOT-5019 3c	
K/A System:	233000 Fuel Pool Cooling/Cleanup	Importance; RO 2.8
K/A Statement:	K4.03 - Knowledge of Fuel Pool Cooling/Cleanup design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool temperature	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

31 **ID: 1659126** **Points: 1.00**

Unit 2 is operating at 100% power when RPS Bus B de-energizes due to a sustained electrical fault.

Which one of the following identifies a radiation monitor that is affected by this electrical loss and how the radiation monitor fails?

A. Main Stack; fails low

B. Main Stack; fails as-is

C. Main Steam Line; fails low

D. Main Steam Line; fails as-is

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	RPS Bus B supplies power to Main Steam Line radiation monitors. On loss of power, these radiation monitors fail low.
Distracters:	A	Main Stack radiation monitors are powered from 20Y050 and 20Y034 (UPS powered panels), not the RPS Buses. Plausible because these are similar radiation monitors and their power supply is also 120 VAC.
	B	Main Stack radiation monitors are powered from 20Y050 and 20Y034 (UPS powered panels), not the RPS Buses. Plausible because these are similar radiation monitors and their power supply is also 120 VAC. The radiation monitors fail low, not as-is. Plausible because some other radiation monitors fail as-is on loss of certain power supplies (ex. RBCCW on loss of 30Y033).
	D	The Main Steam Line radiation monitors fail low, not as-is. Plausible because some other radiation monitors fail as-is on loss of certain power supplies (ex. RBCCW on loss of 30Y033).

Question 31 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659126
User-Defined ID:	
Cross Reference Number:	
Topic:	RPS Bus B loss affect on MSL radiation monitors
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(11)

Source Documentation

Source:	New Exam item **X** Modified Bank **(1139586)** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	M-1-S-26, ARC 218 D-3, PLOT-5063	
Learning Objective:	PLOT-5063-7a	
K/A System:	272000 Radiation Monitoring	Importance; RO 3.0
K/A Statement:	K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM: Reactor protection system	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

32 **ID: 1601380** **Points: 1.00**

A Unit 2 plant startup is in progress with the following:

- Reactor pressure is 500 psig.
- The in-service Control Rod Drive flow control valve fails closed.

Based on this information, Control Rods __(1)__ be moved with the RMCS and Control Rods __(2)__ insert on a Scram.

A. (1) can
(2) will

B. (1) can
(2) will NOT

C. (1) can NOT
(2) will

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

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	Closure of the in-service CRD FCV blocks flow to the drive water and cooling water headers, but not the charging water header. The ability to drive rods with RMCS is lost, but charging water header pressure is still available to ensure the scram function will work. Based on the low initial Reactor pressure, if charging water header pressure was not maintained, the scram function may not be assured and a Reactor scram would be required.
Distracters:	A	The ability to drive rods with RMCS is lost. Plausible because not all CRD hydraulic functions are lost (scram is still assured). Also plausible if candidate does not correctly recall location of CRD FCV relative to drive water header.
	B	The ability to drive rods with RMCS is lost. Plausible because not all CRD hydraulic functions are lost (scram is still assured). Also plausible if candidate does not correctly recall location of CRD FCV relative to drive water header. The Reactor scram function is still assured. Plausible because based on the low initial Reactor pressure, if charging water header pressure was not maintained, the scram function may not be assured.
	D	The Reactor scram function is still assured. Plausible because based on the low initial Reactor pressure, if charging water header pressure was not maintained, the scram function may not be assured.

Question 32 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601380
User-Defined ID:	
Cross Reference Number:	
Topic:	Effect of CRD FCV failure on scram and RMCS
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	**X** New Exam item Modified Bank **(JAF 4/14 NRC #38)** Exam Bank ILT Exam Bank	Previous NRC Exam Other
Reference(s):	M-356	
Learning Objective:	PLOT-5003 7a	
K/A System:	201003 Control Rod and Drive Mechanism	Importance; RO 3.3
K/A Statement:	K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the Control Rod and Drive Mechanism: Control rod drive hydraulic system	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

33 **ID: 993550** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Fire header pressure drops to 115 psig.
- Fire header pressure returns to 150 psig.

Which one of the following (1) describes the status of the Motor Driven and Diesel Driven Fire pumps and (2) the location from which the running Fire pump(s) are shut down?

A. (1) Only the Motor Driven Fire pump is running.
(2) It is shutdown from the Control Room.

B. (1) Only the Motor Driven Fire pump is running.
(2) It is shutdown locally at the pump.

C. (1) Both the Motor Driven Fire pump and the Diesel Driven Fire pump are running.
(2) They are shutdown from the Control Room.

D. (1) Both the Motor Driven Fire pump and the Diesel Driven Fire pump are running.
(2) They are shutdown locally at the pumps.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The Motor Driven Fire pump is running because fire header pressure lowered below 140 psig. The Diesel Driven Fire pump is also running because fire header pressure lowered below 130 psig. These pumps both must be shutdown locally at the pumps.
Distracters:	A	The Diesel Driven Fire pump is also running because fire header pressure lowered below 130 psig. Plausible because the Motor Driven Fire pump starts first and under different circumstances could restore pressure before start of the Diesel Driven Fire pump. These pumps both must be shutdown locally at the pumps. Plausible that the controls for shutting down the pumps would be in the Control Room on the same panel that contains their annunciators.
	B	The Diesel Driven Fire pump is also running because fire header pressure lowered below 130 psig. Plausible because the Motor Driven Fire pump starts first and under different circumstances could restore pressure before start of the Diesel Driven Fire pump.
	C	These pumps both must be shutdown locally at the pumps. Plausible that the controls for shutting down the pumps would be in the Control Room on the same panel that contains their annunciators.

Question 33 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	993550
User-Defined ID:	ILT-5037-4B-001
Cross Reference Number:	
Topic:	ILT-5037-4b-001 The reactor is operating at 100% power. All fire pumps are operable
Num Field 1:	4060
Num Field 2:	A NRC
Text Field:	A

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(4)

Source Documentation

Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(993550)**	Previous NRC Exam Other Exam
Reference(s):	ARC-201 A-3, E-212 SH 2, SO 37B.2.A	
Learning Objective:	PLOT-5037 3b	
K/A System:	286000 Fire Protection	Importance; RO 2.9
K/A Statement:	A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the Fire Protection controls including: System pressure	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

34 **ID: 1601428** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Drywell cooling performance is degraded.
- Drywell temperature is 141°F and slowly rising.
- Drywell fans are operating as follows:

Drywell Cooling Unit	A Fan	B Fan
A	Running	Standby
B	Running	Standby
C	Running	Standby
D	Running	Standby
E	Running	Standby
F	Running	Standby
G	Running	Standby

Drywell Recirculation Fans	
2AV94-A	Running
2AV94-B	Standby
2BV94-A	Running
2BV94-B	Standby

Which one of the following describes the need to enter Tech Spec LCO 3.6.1.4, Drywell Air Temperature, and the ability to start additional fans per SO 40C.1.A-2, Drywell Ventilation System Startup and Normal Operations?

Tech Spec LCO 3.6.1.4 entry is...

A. required. Additional fans may be started.

B. required. NO additional fans may be started.

C. NOT required. Additional fans may be started.

D. NOT required. NO additional fans may be started.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	Drywell temperature is above the 137°F limit in SO 40C.1.A-2 and the 140°F entry condition on ON-112, but below the LCO limit of TS 3.6.1.4 (145°F). Therefore, TS 3.6.1.4 LCO entry is not required. SO 40C.1.A-2 allows start of additional fans.
Distracters:	A	TS 3.6.1.4 LCO entry is not required because temperature is below 145°F. Plausible because temperature is above the 137°F limit in SO 40C.1.A-2 and the 140°F entry condition on ON-112.
	B	TS 3.6.1.4 LCO entry is not required because temperature is below 145°F. Plausible because temperature is above the 137°F limit in SO 40C.1.A-2 and the 140°F entry condition on ON-112. SO 40C.1.A-2 allows start of additional fans. Plausible because the given setup is the base alignment that procedure normally requires and some systems have limits on simultaneous running of A/B components.
	D	SO 40C.1.A-2 allows start of additional fans. Plausible because the given setup is the base alignment that procedure normally requires and some systems have limits on simultaneous running of A/B components.

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601428
User-Defined ID:	
Cross Reference Number:	
Topic:	TS 3.6.1.4 entry and start of additional DW fans
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (9)

Source Documentation		
Source:	New Exam item **X** Modified Bank **(2008 NRC #52)** ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	TS 3.6.1.4, SO 40C.1.A-2	
Learning Objective:	PLOT-5040C 10c	
K/A System:	223001 Primary CTMT and Aux.	Importance; RO 3.6
K/A Statement:	A2.10 - Ability to (a) predict the impacts of the following on the Primary CTMT and Aux.; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High drywell temperature	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

35 **ID: 1601430** **Points: 1.00**

Unit 2 has experienced a leak in the Drywell with the following:

- Drywell pressure is 3 psig and rising slowly.
- Reactor water level is -30" and rising slowly.
- Reactor pressure is 700 psig and lowering slowly.
- The A loop of RHR is placed in Torus Cooling using RRC 10.1-2, "RHR System Torus Cooling During a Plant Event".

Then, the leak worsens and the following conditions exist:

- Drywell pressure is 20 psig and rising slowly.
- Reactor water level is -110" and lowering slowly.
- Reactor pressure is 400 psig and lowering slowly.

Which one of the following describes the response of the Torus Cooling lineup?

MO-039A, Torus Hdr __(1)__.
MO-034A, Full Flow Test __(2)__.

A. (1) automatically closes
(2) automatically closes

B. (1) automatically closes
(2) remains open

C. (1) remains open
(2) automatically closes

D. (1) remains open
(2) remains open

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	Since initial conditions had Reactor water level above -226", the S18A keylock switch was not operated during initial alignment of Torus Cooling. Therefore, when both Drywell pressure is above 2 psig and Reactor pressure is less than 450 psig, a LPCI signal is received which closes both MO-039A and MO-034A.
Distracters:	B	Both valves automatically close. Plausible because Reactor water level remains above the -160" setpoint for LPCI and the S18A keylock switch would have been operated in different initial conditions. Plausible that only MO-039A would close to isolate both Torus spray and cooling.
	C	Both valves automatically close. Plausible because Reactor water level remains above the -160" setpoint for LPCI and the S18A keylock switch would have been operated in different initial conditions. Plausible that only MO-034A would close to isolate Torus cooling.
	D	Both valves automatically close. Plausible because Reactor water level remains above the -160" setpoint for LPCI and the S18A keylock switch would have been operated in different initial conditions.

Question 35 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601430
User-Defined ID:	
Cross Reference Number:	
Topic:	Torus Cooling response to hi DW press and low RX press
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (8)

Source Documentation

Source:	New Exam item **X** Modified Bank **(2009 NRC #35)** ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	RRC 10.1-2, M-1-S-65	
Learning Objective:	PLOT-5010 3a	
K/A System:	219000 RHR/LPCI: Torus/Pool Cooling Mode	Importance; RO 3.3
K/A Statement:	A3.01 - Ability to monitor automatic operations of the RHR/LPCI: Torus/Pool Cooling Mode including: Valve operation	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

36 **ID: 1659151** **Points: 1.00**

Refueling is in progress on Unit 2 with the following:

- A new fuel bundle is being loaded in the general vicinity of the A WRNM, but NOT directly adjacent to the A WRNM.
- Significant WRNM response is observed.
- ON-124, Fuel Floor and Fuel Handling Problems, is entered.
- The initial A WRNM count was 1.0×10^3 cps.
- The current A WRNM count is 4.2×10^3 cps.
- The current A WRNM period is 100 seconds.
- The fuel bundle has been lowered halfway into the core.

Which one of the following describes the required control of the fuel bundle in accordance with ON-124, Fuel Floor and Fuel Handling Problems?

The fuel bundle...

A. may continue to be lowered as long as counts stay below 8.0×10^3 cps and period stays above 28 seconds.

B. must be held in the current position until period begins to lengthen and Reactor Engineering is consulted.

C. must be raised until period begins to lengthen and then must be held in this new position.

D. must be fully raised from the core.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	If WRNM count rate doubles two times and the fuel assembly is still grappled, then ON-124 requires fully raising the bundle from the core.
Distracters:	A	If WRNM count rate doubles two times and the fuel assembly is still grappled, then ON-124 requires fully raising the bundle from the core. Plausible because WRNM response has just cleared the threshold for ON-124 entry, so understandable that continued lowering would be allowed with increased monitoring until a further threshold is reached. 8.0×10^3 cps would be one additional count rate doubling and 28 cps is the short period rod block setting.
	B	If WRNM count rate doubles two times and the fuel assembly is still grappled, then ON-124 requires fully raising the bundle from the core. Plausible because holding in the current position stops all reactivity change, WRNM response has just cleared the threshold for ON-124 entry, and period is not fast enough to cause the short period alarm yet.
	C	If WRNM count rate doubles two times and the fuel assembly is still grappled, then ON-124 requires fully raising the bundle from the core. Plausible because raising the fuel bundle inserts negative reactivity to turn period and then holding the fuel bundle where it is minimizes any further chance of a transient.

Question 36 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659151
User-Defined ID:	
Cross Reference Number:	
Topic:	ON-124 actions based on WRNM indications
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (13)

Source Documentation		
Source:	**X** New Exam item Exam Modified Bank Bank ILT Exam Bank	Previous NRC Other Exam
Reference(s):	ON-124	
Learning Objective:	PLOT-1550 27a	
K/A System:	234000 Fuel Handling Equipment	Importance; RO 3.7
K/A Statement:	A4.01 - Ability to manually operate and/or monitor in the control room: Neutron monitoring system	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

37 **ID: 1601438** **Points: 1.00**

Unit 2 is operating at 100% power and the Reactor pressure indication on SPDS is currently displayed in cyan.

Which one of the following describes what this indication means regarding control of Reactor pressure by EHC?

EHC...

A. is controlling Reactor pressure within all SPDS limits.

B. is controlling Reactor pressure above or below the SPDS warning limit, but within the SPDS alarm limit.

C. is controlling Reactor pressure above or below the SPDS warning and alarm limits.

D. performance CANNOT be reliably determined by this indication because there is a problem with the computer point.

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	A cyan indication on SPDS indicates that the computer point is OK and within both the warning and alarm limits.
Distracters:	B	A cyan indication on SPDS indicates that the computer point is OK and within both the warning and alarm limits. Plausible because this would be correct if the indication was yellow.
	C	A cyan indication on SPDS indicates that the computer point is OK and within both the warning and alarm limits. Plausible because this would be correct if the indication was red.
	D	A cyan indication on SPDS indicates that the computer point is OK and within both the warning and alarm limits. Plausible because this would be correct if the indication was magenta.

Question 37 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601438
User-Defined ID:	
Cross Reference Number:	
Topic:	SPDS Reactor pressure indication - cyan
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(7)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	PLOT-5059K	
Learning Objective:	PLOT-5001D 11	
K/A System:	241000 Reactor/Turbine Pressure Regulator	Importance; RO 3.9
K/A Statement:	2.1.19 - Ability to use plant computers to evaluate system or component status.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

38 **ID: 1650248** **Points: 1.00**

A Unit 2 Reactor shutdown is in progress with the following:

- Reactor power is below the point of adding heat.
- The Rod Worth Minimizer (RWM) is latched on Group 6
- Group 6 contains 12 control rods that are to be inserted from position 12 to position 08.
- The first control rod inserted triple notched to position 04.

Which one of the following describes the status of RWM rod blocks with the control rod at position 04 and the response of the insert error once the control rod is returned to position 08?

A RWM rod block ___(1)___ be enforced.
Once the control rod is returned to notch 08, the insert error ___(2)___.

A. (1) will
(2) will automatically clear

B. (1) will
(2) must be manually reset

C. (1) will NOT
(2) will automatically clear

D. (1) will NOT
(2) must be manually reset

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	Two insert errors are required to initiate a RWM rod block, therefore this single insert error does not result in a RWM rod block. The RWM insert error indication automatically updates as new control rod position information is provided, therefore the insert error automatically clears once the control rod is returned to position 08.
Distracters:	A	Two insert errors are required to initiate a RWM rod block, therefore this single insert error does not result in a RWM rod block. Plausible because a single withdraw error results in a RWM rod block.
	B	Two insert errors are required to initiate a RWM rod block, therefore this single insert error does not result in a RWM rod block. Plausible because a single withdraw error results in a RWM rod block. The RWM insert error indication automatically updates as new control rod position information is provided, therefore the insert error automatically clears once the control rod is returned to position 08. Plausible because some alarms require manual reset, such as a control rod drift alarm.
	D	The RWM insert error indication automatically updates as new control rod position information is provided, therefore the insert error automatically clears once the control rod is returned to position 08. Plausible because some alarms require manual reset, such as a control rod drift alarm.

Question 38 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650248
User-Defined ID:	
Cross Reference Number:	
Topic:	Insert error and rod block
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(7)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC-211 F-5, PLOT-5062A	
Learning Objective:	PLOT-5062A 4j	
K/A System:	201006 RWM	Importance; RO 3.2
K/A Statement:	A4.05 - Ability to manually operate and/or monitor in the control room: Rod insert error indication: P-Spec (Not-BWR6)	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

39 **ID: 1104790** **Points: 1.00**

Unit 3 has experienced an accident with the following:

- The crew is executing T-111, Level Restoration.
- Reactor water level is -190" and slowly lowering.
- Reactor pressure is 550 psig and slowly lowering.
- An emergency depressurization is in progress with 5 SRVs open.
- The 3A Core Spray pump is the only available source of injection.

Which one of the following describes the status of Adequate Core Cooling (ACC) in accordance with the Emergency Operating Procedures?

Adequate Core Cooling (ACC) is...

A. NOT being maintained.

B. being maintained by submergence.

C. being maintained by spray cooling.

D. being maintained by steam cooling.

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	ACC is being maintained by steam cooling because Reactor water level is below -172" (minimum level for submergence) but above -195" (minimum level for steam cooling).
Distracters:	A	ACC is being maintained by steam cooling because Reactor water level is below -172" (minimum level for submergence) but above -195" (minimum level for steam cooling). Plausible because if Reactor water level goes below -195", ACC will not be maintained.
	B	Submergence is not maintained because Reactor water level is below -172". Plausible because Reactor water level is above other values associated with ACC (-195", -205", -226").
	C	Spray cooling is not available because only one Core Spray pump is available (two needed for proper loop flow). Plausible because Reactor water level is above -226" and some Core Spray flow is available.

Question 39 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	1104790
User-Defined ID:	ILT-2111-4-002
Cross Reference Number:	295031 EA2.04
Topic:	ILT-2111-4-002 The following conditions exist on Unit 3: *The crew is executing T-111 "Level Restora
Num Field 1:	
Num Field 2:	A NRC
Text Field:	NRC-09-1

Comments:

Psychometrics

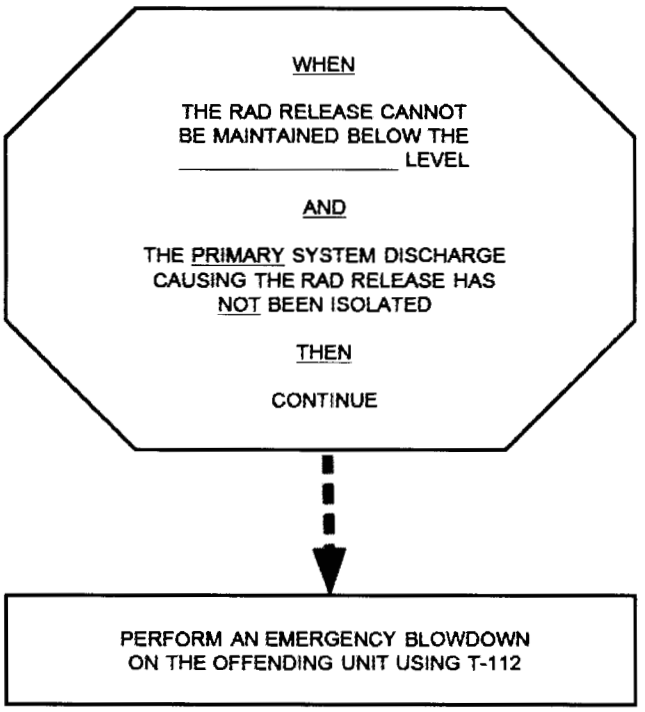
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (10)

Source Documentation

Source:	New Exam item **(2011 NRC #52)** **X** Previous NRC Exam Modified Bank Other Exam Bank ILT Exam Bank	
Reference(s):	T-111 and Bases	
Learning Objective:	PLOT-PBIG-2111 4	
K/A System:	295031 Reactor Low Water Level	Importance; RO 4.6
K/A Statement:	K1.01 - Knowledge of the operational implications of the following concepts as they apply to Reactor Low Water Level: Adequate core cooling	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

40 **ID: 1650233** **Points: 1.00**

T-104, Radioactivity Release Control, contains the following steps:



Which one of the following identifies (1) the Emergency Action Level that correctly fills in the blank and (2) the reason for the Emergency Blowdown, in accordance with T-104?

A. (1) Alert
(2) Limit dose to the general public

B. (1) Alert
(2) Maintain Control Room habitability

C. (1) General Emergency
(2) Limit dose to the general public

D. (1) General Emergency
(2) Maintain Control Room habitability

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	T-104 requires Emergency Blowdown when the rad release rate exceeds the General Emergency EAL threshold. The basis is to limit dose to the general public.
Distracters:	A	The General Emergency is the correct EAL threshold, not the Alert. Plausible because the Alert level is used in T-104 to determine when to scram and isolate systems.
	B	The General Emergency is the correct EAL threshold, not the Alert. Plausible because the Alert level is used in T-104 to determine when to scram and isolate systems. The basis is to limit dose to the general public, not to maintain Control Room habitability. Plausible because Control Room habitability is a concern during a release and is part of the design basis for the Control Room and Control Room HVAC.
	D	The basis is to limit dose to the general public, not to maintain Control Room habitability. Plausible because Control Room habitability is a concern during a release and is part of the design basis for the Control Room and Control Room HVAC.

Question 40 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1650233
User-Defined ID:	
Cross Reference Number:	
Topic:	When to Blowdown in T-104
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation		
Source:	New Exam item **(SSES LOC26R #45)** X Previous NRC Exam Modified Bank Other Exam Bank ILT Exam Bank	
Reference(s):	T-104 and bases	
Learning Objective:	PLOT-PBIG-2104 5	
K/A System:	295038 High Off-site Release Rate	Importance; RO 4.2
K/A Statement:	K1.02 - Knowledge of the operational implications of the following concepts as they apply to High Off-site Release Rate: Protection of the general public	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

41 **ID: 1601471** **Points: 1.00**

Which one of the following identifies the minimum Torus water level for operation of HPCI and the reason for tripping HPCI if Torus water level CANNOT be maintained above this level in accordance with T-102, Primary Containment Control?

A. 6' to prevent directly pressurizing the Torus

B. 6' to prevent vortex damage to the HPCI pump

C. 9.5' to prevent directly pressurizing the Torus

D. 9.5' to prevent vortex damage to the HPCI pump

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	T-102 steps T/L-12 and 13 secure HPCI if Torus water level cannot be maintained above 9.5'. This is based on when the HPCI exhaust line becomes uncovered, which would result in directly pressurizing the Torus if HPCI operation continued.
Distracters:	A	The level is 9.5', not 6'. Plausible because RCIC is secured at 6'.
	B	The level is 9.5', not 6'. Plausible because RCIC is secured at 6'. The reason is to prevent directly pressurizing the Torus, not to prevent HPCI pump damage. Plausible because HPCI does have a vortex limit and RCIC is secured at 6' for this reason.
	D	The reason is to prevent directly pressurizing the Torus, not to prevent HPCI pump damage. Plausible because HPCI does have a vortex limit and RCIC is secured at 6' for this reason.

Question 41 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601471
User-Defined ID:	
Cross Reference Number:	
Topic:	HPCI trip on low Torus level and bases
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank Other Exam Bank ILT Exam Bank	
Reference(s):	T-102 and bases	
Learning Objective:	PLOT-PBIG-2102 6	
K/A System:	295030 Low Suppression Pool Wtr Lvl	Importance; RO 3.8
K/A Statement:	K1.01 - Knowledge of the operational implications of the following concepts as they apply to Low Suppression Pool Wtr Lvl: Steam condensation	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

42 **ID: 1601472** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- The voltage on the normal offsite feeder for the E-12 bus degrades to 89% of rated voltage and remains steady at that value.

Which one of the following describes the resulting status of the E-12 Bus and Group II isolation valves two (2) minutes later?

The E-12 Bus is energized from the __(1)__ and __(2)__ Group II isolation valves have re-positioned.

A. (1) alternate offsite feed
(2) NO

B. (1) E-1 Diesel Generator
(2) NO

C. (1) alternate offsite feed
(2) some

D. (1) E-1 Diesel Generator
(2) some

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	Bus E-12 automatically transfers to the alternate feeder breaker after approximately 61 seconds (<99.8% of rated voltage without LOCA signal present). The E-124 load center supply breaker opens on the load shed and results in a loss of 20Y033 panel and a subsequent Inboard Group II isolation due to the power loss of PCIS relays. This half isolation does re-position inboard valves. **Note:** The question meets the K/A because it presents an electrical grid disturbance (degraded voltage on offsite feeder) and requires the candidate to understand how this interrelates with breaker control for a specific plant bus (Bus E-12 normal/alternate feeder breaker control).
Distracters:	A	An inboard isolation occurs, which does re-position some Group II isolation valves. Plausible if candidate mixes up which electrical supply loss causes which isolation or thinks the half isolation does not re-position any valves.
	B	Bus E-12 automatically transfers to the alternate feeder breaker, not the EDG. Plausible because the EDG could also energize the bus under different conditions. An inboard isolation occurs, which does re-position some Group II isolation valves. Plausible if candidate mixes up which electrical supply loss causes which isolation or thinks the half isolation does not re-position any valves.
	D	Bus E-12 automatically transfers to the alternate feeder breaker, not the EDG. Plausible because the EDG could also energize the bus under different conditions.

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601472
User-Defined ID:	
Cross Reference Number:	
Topic:	E-12 undervoltage
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(4)

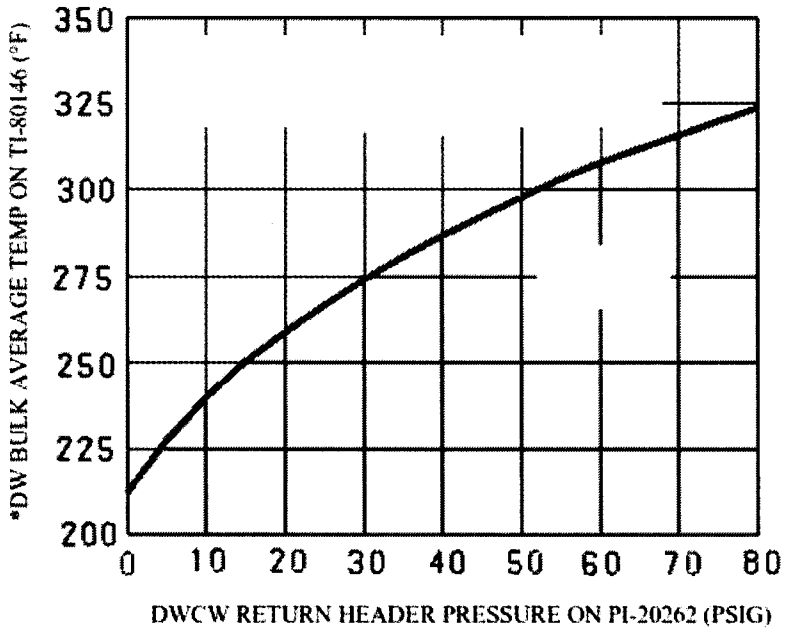
Source Documentation

Source:	New Exam item **(2008 NRC #18)**	**X** Previous NRC Exam
	Modified Bank	Other Exam
	Bank	
	ILT Exam Bank	
Reference(s):	ARC-001 D-1, SO 54.7.A	
Learning Objective:	PLOT-5054 7b	
K/A System:	700000 Generator Voltage and Electric Grid Disturbances	Importance; RO 3.1
K/A Statement:	K2.02 - Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Breakers, relays	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:	TRH 9/13/16 - Could change 2nd half to ask if any isolation valves re-position or not.	

43 **ID: 994781** **Points: 1.00**

Unit 2 has experienced a coolant leak with the following:

- Drywell pressure is 6 psig and stable.
- Drywell sprays are not available.
- Drywell bulk average temperature is 275°F and stable.
- Drywell coolers are being supplied by Drywell Chilled Water (DWCW).
- DWCW return header pressure is 28 psig.
- The Reactor and Radwaste Buildings are not accessible.
- T-223-2, Drywell Cooler Fan Bypass, is being implemented.



Which one of the following describes the ability to restart Drywell Cooler fans in accordance with T-223-2?

Drywell Cooler fans...

A. may be restarted in Slow speed only.

B. may be restarted in Slow or Fast speed.

C. CANNOT be restarted until Reactor Building access is restored.

D. CANNOT be restarted until an Engineering evaluation is obtained.

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	Drywell cooler fan units cannot be restarted since operation of the coolers plots on the UNSAFE side of the Drywell Chilled Water Saturation Curve (T-223 Figure 1). Per step 4.1 of T-223, operation must be verified to be on the safe side of the curve, or an engineering evaluation must be obtained, prior to starting (or restarting) any drywell cooler fan unit.
Distracters:	A	Drywell cooler fan units cannot be restarted. Plausible since T-223 states the drywell cooler fans should be started in SLOW speed if drywell pressure is above 0.75 psig.
	B	Drywell cooler fan units cannot be restarted. Plausible since T-223 states the drywell cooler fans should be started in SLOW speed if drywell pressure is above 0.75 psig, but allows starting the fans in FAST speed if the local fan speed control switches are not accessible (as is the case here).
	C	Drywell cooler fan units cannot be restarted. Plausible since the local fan speed control switches are not accessible.

Question 43 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	994781
User-Defined ID:	ILT-1560-4-002
Cross Reference Number:	295028 EA2.01
Topic:	ILT-1560-4-002 Unit 2 was operating at 100% powe when a Loss of Coolant Accident occurred. The
Num Field 1:	
Num Field 2:	A NRC
Text Field:	NRC-09-1

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (10)

Source Documentation

Source:	New Exam item **(2011 NRC #51)** **X** Previous NRC Exam Modified Bank Other Exam Bank **X** ILT Exam Bank	
Reference(s):	T-223-2	
Learning Objective:	PLOT-1560 4	
K/A System:	295028 High Drywell Temperature	Importance; RO 3.2
K/A Statement:	K2.02 - Knowledge of the interrelations between High Drywell Temperature and the following: Components internal to the drywell	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

44 **ID: 994756** **Points: 1.00**

Both Units are operating at 100% power with the following:

- Annunciator 216 C-4, INSTRUMENT AIR DRYER TROUBLE, alarms.
- Annunciator 216 D-4, B INSTRUMENT AIR HEADER LO PRESS, alarms.
- B Instrument Air Header Pressure (PI-2425B) on Panel 20C012 is lowering.
- B Instrument Air Receiver Pressure (PI-2429B) on Panel 20C012 is steady at 110 psig.
- The Equipment Operator reports there is a valve malfunction on the Unit 2 B Instrument Air Dryer.

Which one of the following describes the required action to restore instrument air header pressure?

A. Bypass the Unit 2 B Instrument Air Dryer.

B. Cross-tie the Unit 2 A and B instrument air headers.

C. Cross-tie the Unit 2 and Unit 3 B instrument air headers.

D. Cross-tie the Unit 2 service air and B instrument air headers.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	The given conditions indicate both towers for the B air dryer are isolated, which means there is no flow to the B instrument air header from the B air compressor/receiver. Therefore, B instrument air header pressure will continue to lower. Per ON-119, the correct action to take for this condition is to cross-tie the Unit 2 and Unit 3 B instrument air headers.
Distracters:	A	There is no provision for bypassing a malfunctioning dryer in ON-119 or ARC-216 C-4. Both references direct cross-tying the Unit 2 and Unit 3 instrument air headers. Plausible because if bypass was available, it would restore header pressure.
	B	Cross-tying the A and B instrument air headers will not be effective in restoring B instrument air header pressure since the A supply must pass through the B air dryer in order to supply the B header. Plausible because this would be correct for a loss of compressor or if the crosstie point were downstream of the air dryers
	D	Cross-tying the service air and B instrument air headers will not be effective in restoring B instrument air header pressure since the service air supply must pass through the B air dryer in order to supply the B header. Plausible because this would be correct for a loss of compressor or if the crosstie point were downstream of the air dryers.

Question 44 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	2.00
System ID:	994756
User-Defined ID:	ILT-5036-7B-002
Cross Reference Number:	300000 K3.03
Topic:	ILT-5036-7B-002 The Instrument Air System is in a normal lineup when the following occur: *Instrume
Num Field 1:	
Num Field 2:	A NRC
Text Field:	NRC-09-1

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(4)

Source Documentation

Source:	New Exam item **(2011 NRC #23)** Modified Bank Bank **X** ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	ARC-216 C-4, ON-119, M-320	
Learning Objective:	PLOT-5036 7b	
K/A System:	295019 Partial or Total Loss of Inst. Air	Importance; RO 3.2
K/A Statement:	K2.14 - Knowledge of the interrelations between Partial or Total Loss of Inst. Air and the following: Plant air systems	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

45 **ID: 1601508** **Points: 1.00**

Per AO 2A.1-2, Recirculation System Single Loop Operation, indicated core flow must be corrected (calculated) IF operating Recirc Pump speed is >650 RPM AND Indicated Core Flow is >35 Mlbs/hr.

Which one of the following describes the reason for this requirement?

The reason for correcting indicated core flow is to account for...

A. stall flow in the idle loop jet pumps.

B. reduced core plate differential pressure.

C. reverse flow through the idle loop jet pumps.

D. forward flow through the idle loop jet pumps.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	Above 650 RPM Recirc pump speed and 35 Mlbs/hr indicated core flow, reverse flow through the idle loop jet pumps results in erroneous indicated core flow. This is accounted for by subtracting ~2 times the idle loop flow.
Distracters:	A	The reason is reverse flow through idle jet pumps. Plausible because stall flow occurs at or near 650 RPM Recirc pump speed and is another concern addressed by AO 2A.1-2 due to vibration issues.
	B	The reason is reverse flow through idle jet pumps. Plausible because core plate d/p impacts Core Plate Flow (which is indicated on the same Control Room recorder), but does not impact Indicated Core Flow.
	D	The reason is reverse flow through idle jet pumps. Plausible because forward flow through the idle loop jet pumps occurs below 650 RPM Recirc pump speed or 35 Mlbs/hr indicated core flow.

Question 45 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601508
User-Defined ID:	
Cross Reference Number:	
Topic:	Reason for indicated core flow correction in single loop
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(2)

Source Documentation

Source:	New Exam item **(2009 NRC #47)** Modified Bank Bank **X** ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	AO 2A.1-2; GP-5	
Learning Objective:	PLOT-5002 4b	
K/A System:	295001 Partial or Complete Loss of Forced Core Flow Circulation	Importance; RO 2.9
K/A Statement:	K3.06 - Knowledge of the reasons for the following responses as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Core flow indication	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

46 **ID: 1601528** **Points: 1.00**

Which one of the following describes the reason why the Reactors are manually scrammed and the Main Turbines are tripped prior to evacuating the Main Control Room, in accordance with SE-1, Plant Shutdown from the Remote Shutdown Panel?

A. Ensures Reactor water level can be maintained by operating HPCI from the Remote Shutdown Panels.

B. Ensures Reactor water level can be maintained by operating RCIC from the Remote Shutdown Panels.

C. Taking these actions from outside the Main Control Room would require access to plant areas that may be inaccessible due to fire.

D. Taking these actions from outside the Main Control Room would require access to plant areas that may be inaccessible due to post-accident high radiation levels.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	SE-1 bases state that scramming the Reactor and verifying the Main Turbine is tripped "puts the plant in a condition that will allow maintaining safe shutdown from the Remote Shutdown Panels". This includes control of Reactor water level from the Remote Shutdown Panels using RCIC.
Distracters:	A	The Control Room evacuation strategy is based on maintaining Reactor water level with RCIC from the RSPs, not HPCI. Plausible because the similar SE-10 procedure, Alternate Shutdown, utilizes HPCI for level control.
	C	The basis is not accessability of local controls due to fire. Plausible because a Control Room abandonment is likely caused by fire in the Control Room.
	D	The Control Room evacuation analysis does not assume simultaneous accident conditions. Plausible because if high radiation conditions did exist in the plant, actions to scram the Reactor and/or trip the Main Turbine in the field could be affected.

Question 46 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601528
User-Defined ID:	
Cross Reference Number:	
Topic:	SE-1 basis for scramming and turbine trip
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item NRC Exam **(SSES LOC27 NRC #45)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	SE-1 and bases	
Learning Objective:	PLOT-1555 9	
K/A System:	295016 Control Room Abandonment	Importance; RO 3.7
K/A Statement:	K3.02 - Knowledge of the reasons for the following responses as they apply to Control Room Abandonment: Turbine trip	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

47 **ID: 1601537** **Points: 1.00**

Unit 2 is operating at 100% power when Reactor pressure rises to 1120 psig.

Which one of the following describes the response of the Reactor Recirculation pumps (RRP) and the reason for this response?

The RRPs will be...

A. tripped to reduce risk of RRP seal failure.

B. running at 30% speed to reduce risk of RRP seal failure.

C. tripped to rapidly reduce Reactor power using Recirculation flow.

D. running at 30% speed to rapidly reduce Reactor power using Recirculation flow.

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	Reactor pressure >1106 psig trips the RRPs. This trip is designed to rapidly reduce Reactor power using Recirc flow.
Distracters:	A	The reason is to rapidly reduce Reactor power. Plausible because high Reactor pressure does lead to increased stresses on the Recirc pump seals which increase the risk of seal failure. Tripping the pump is a plausible method for reducing stresses on the seals from pump operation.
	B	RRPs are tripped, not just runback to 30% speed. Plausible because multiple conditions do runback RRPs to 30% speed and this would reduce operating stresses on seals. The reason is to rapidly reduce Reactor power. Plausible because high Reactor pressure does lead to increased stresses on the Recirc pump seals which increase the risk of seal failure. Lowering pump speed is a plausible method for reducing stresses on the seals from pump operation.
	D	RRPs are tripped, not just runback to 30% speed. Plausible because multiple conditions do runback RRPs to 30% speed and this would reduce Reactor power rapidly.

Question 47 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601537
User-Defined ID:	
Cross Reference Number:	
Topic:	High Reactor pressure trip and reason
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	New Exam item Modified Bank Bank **(NMP1 2013 Cert #12)** ILT Exam Bank	Previous NRC Exam **X** Other Exam
Reference(s):	ARC-214 A-3, DBD P-S-19	
Learning Objective:	PLOT-5002 4e	
K/A System:	295025 High Reactor Pressure	Importance; RO 3.9
K/A Statement:	K3.02 - Knowledge of the reasons for the following responses as they apply to High Reactor Pressure: Recirculation pump trip: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

48 **ID: 995506** **Points: 1.00**

Unit 2 is operating at 100% power when a spurious Main Turbine trip and resulting Generator lockout occurs.

Which one of the following describes the status of 13 KV Aux Bus load circuit breakers immediately following the Generator lockout?

ASSUMING NO OPERATOR ACTIONS, immediately following the Generator lockout, all 13 KV Aux Bus load circuit breakers are...

A. open as part of the load shed.

B. closed with power to the loads.

C. open, except those supplying 480 VAC load centers.

D. closed, except the Recirc ASD breakers.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	All of the 13 KV breakers are closed except for the Recirc ASD breakers. All 13 KV loads remain energized following the fast transfer. Plausible if the candidate doesn't understand the fast transfer logic or confuses the 13KV load breakers with the loads (cooling tower lift pumps and fans if powered from the same S/U source and stator cooling pump breakers) that load shed.
Distracters:	A	All of the 13 KV breakers are closed except for the Recirc ASD breakers. All 13 KV loads remain energized following the fast transfer. Plausible if the candidate doesn't understand the fast transfer logic or confuses the 13KV load breakers with the loads (cooling tower lift pumps and fans if powered from the same S/U source and stator cooling pump breakers) that load shed.
	B	The Recirc ASD breakers do load shed. Plausible if the candidate does not recall the components that load shed or forgets that the Recirc ASD breakers must load shed for LOCA loading.
	C	All of the 13 KV breakers are closed except for the Recirc ASD breakers. All 13 KV loads remain energized following the fast transfer. Plausible if the candidate doesn't understand the fast transfer logic or confuses the 13 KV load breakers with the loads (cooling tower lift pumps and fans if powered from the same S/U source and stator cooling pump breakers) that load shed.

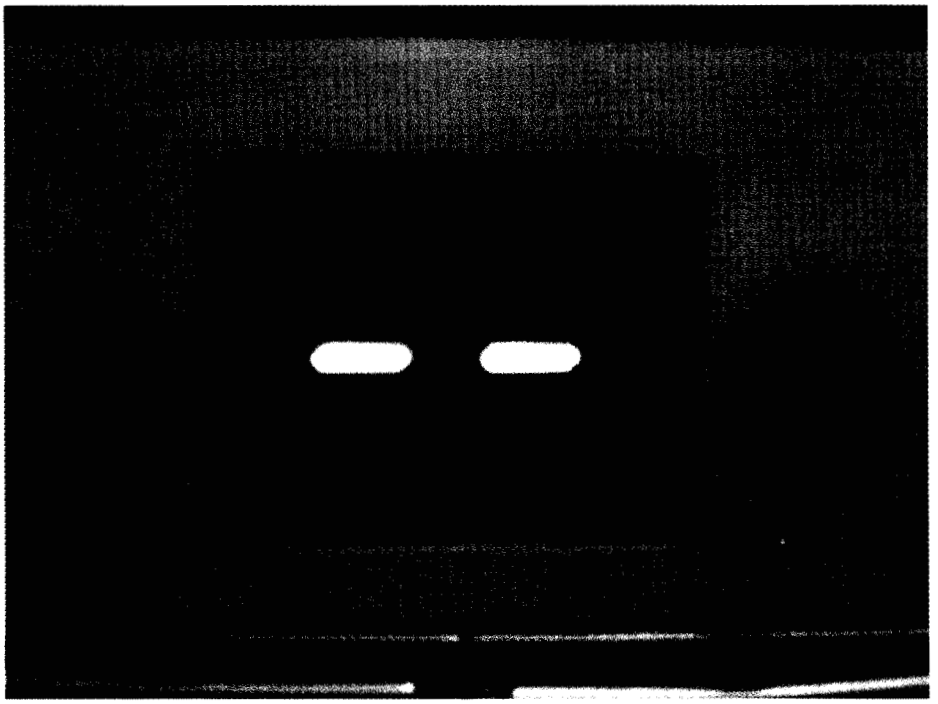
Question 48 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	1.00
System ID:	995506
User-Defined ID:	13 CERT B CERT
Cross Reference Number:	295005 AA1.07
Topic:	ILT 5053-3b-001 B & C CERT
Num Field 1:	C CERT
Num Field 2:	A NRC
Text Field:	

Comments:

Source Documentation		
Level of Difficulty		10CFR55.41(b)(7)
Memory		
Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(995506)**	Previous NRC Exam Other Exam
Reference(s):	E-92, SO 53.7.E	
Learning Objective:	PLOT-5053 3b	
K/A System:	295005 Main Turbine Generator Trip	Importance; RO 3.1
K/A Statement:	A1.01 - Ability to operate and/or monitor the following as they apply to Main Turbine Generator Trip: Recirculation system: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

49 **ID: 1601608** **Points: 1.00**

The following position indication is present for a control rod:



Which one of the following describes this control rod position indication?

This is the control rod position indication for a control rod that is...

A. scrammed, after the scram is reset.

B. scrammed, before the scram is reset.

C. having a coupling check performed and is uncoupled.

D. having a coupling check performed and is properly coupled.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	Double green dashes indicate that the control rod is picking up the XX reed switch. This reed switch is picked up when the control rod is inserted past the latched 00 position, which occurs when the control rod is scrammed with the scram not yet reset.
Distracters:	A	This is the indication for a control rod that is scrammed with the scram not yet reset. Plausible because the control rod is scrammed and indication would also be green after the scram is reset.
	C	This is the indication for a control rod that is scrammed with the scram not yet reset. Plausible because indication would be similar double red dashes during a coupling check with the control rod uncoupled.
	D	This is the indication for a control rod that is scrammed with the scram not yet reset. Plausible because similar double red dashes could be seen during a coupling check, and the green dashes without a "48" could be mistaken as the proper indication with drive pressure applied to a coupled rod during a coupling check (green = good, red = bad).

Question 49 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601608
User-Defined ID:	
Cross Reference Number:	
Topic:	Position indication with scram not reset
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-100, PLOT-5003	
Learning Objective:	PLOT-5003 4i	
K/A System:	295006 SCRAM	Importance; RO 4.1
K/A Statement:	A1.07 - Ability to operate and/or monitor the following as they apply to SCRAM: Control rod position	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

50 **ID: 1601609** **Points: 1.00**

Unit 2 is operating at 100% power when the following occurs:

Time (minutes)	Condition
0	Drywell pressure is 2.1 psig and slowly rising.
2	Annunciator 004 C-1, E32 BUS DIFFERENTIAL OR OVERCURRENT RELAYS, alarms.
5	Annunciator 002 B-5, A EMERG SERVICE WATER PUMP TRIP, alarms.

Which one of the following describes the status of the B ESW pump and the ECW pump at Time 7 minutes?

The B ESW pump is __(1)__. The ECW pump is __(2)__.

A. (1) running
(2) running

B. (1) NOT running
(2) running

C. (1) running
(2) NOT running

D. (1) NOT running
(2) NOT running

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	When drywell pressure rises above 2 psig the A and B ESW pumps and the ECW pump will automatically start. With ESW header pressure > 30 psig, the ECW pump will automatically shutdown 45 seconds after it first started. Since the B ESW pump is powered from the E32 bus, it will trip as a result of the E32 bus differential/overcurrent condition. When the A ESW pump trips, the ECW pump will automatically restart when ESW header pressure drops below 30 psig.
Distracters:	A	The B ESW pump is not running due to the E32 bus differential/overcurrent condition. Plausible if the applicant does not know pump power supplies or recognize the E32 bus is locked out.
	C	The B ESW pump is not running due to the E32 bus differential/overcurrent condition. Plausible if the applicant does not know pump power supplies or recognize the E32 bus is locked out. The ECW pump is running due to the loss of both ESW pumps (low ESW header pressure). Plausible if the applicant does not know ECW pump auto-start conditions or recognize loss of both ESW pumps.
	D	The ECW pump is running due to the loss of both ESW pumps (low ESW header pressure). Plausible if the applicant does not know ECW pump auto-start conditions or recognize loss of both ESW pumps.

Question 50 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601609
User-Defined ID:	
Cross Reference Number:	
Topic:	ECW backup to ESW
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(4)

Source Documentation

Source:	New Exam item **(2011 NRC #26)** Modified Bank Bank **X** ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	ARC 212 B-2, ARC-004 C-1	
Learning Objective:	PLOT-5033 3b	
K/A System:	295018 Partial or Total Loss of CCW	Importance; RO 3.3
K/A Statement:	A1.01 - Ability to operate and/or monitor the following as they apply to Partial or Total Loss of CCW: Backup systems	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

51 **ID: 1601610** **Points: 1.00**

Unit 2 is shutdown with the following:

- A loss of Shutdown Cooling has occurred.
- The following Reactor coolant temperature data has been collected:

Time (minutes)	Reactor Coolant Temperature (°F)
0	140
15	158

Which one of the following identifies (1) if the current heatup rate is above or below the Technical Specification limit and (2) when a mode change would first occur if this heatup rate continued?

A. (1) Above limit
(2) Time 50 minutes

B. (1) Above limit
(2) Time 60 minutes

C. (1) Below limit
(2) Time 50 minutes

D. (1) Below limit
(2) Time 60 minutes

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The data gives a heatup rate of (18°F/15 min*60 min/hr=72°F/hr). This is below the Technical Specification limit of 100°F/hr. A mode change will occur when temperature reaches 212°F, which will occur at Time 60 minutes (140 °F +18°F/15 min*60 min=212°F).
Distracters:	A	A mode change will occur when temperature reaches 212°F, which will occur at Time 60 minutes. 50 minutes is plausible because this is when temperature reaches 200°F, which is the mode change on some other plants.
	B	A mode change will occur when temperature reaches 212°F, which will occur at Time 60 minutes. 50 minutes is plausible because this is when temperature reaches 200°F, which is the mode change on some other plants.
	C	The heatup rate of 72°F/hr is below the applicable TS limit. Plausible because it is above the 20°F/hr limit during hydrostatic testing.

Question 51 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1601610
User-Defined ID:	
Cross Reference Number:	
Topic:	Loss of SDC - heatup rate and mode change
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(3)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	TS Table 1.1-1, TS 3.4.9, PTLR	
Learning Objective:	PLOT-5004 12	
K/A System:	295021 Loss of Shutdown Cooling	Importance; RO 3.5
K/A Statement:	A2.01 - Ability to determine and/or interpret the following as they apply to Loss of Shutdown Cooling: Reactor water heatup/cooldown rate	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

52 **ID: 1659123** **Points: 1.00**

Unit 2 has experienced a failure to scram with the following:

- Reactor power is oscillating between 5 and 10%.
- Torus water temperature is 112°F and rising slowly.

Based on these conditions, which one of the following describes the need for injecting boron in accordance with T-101, RPV Control?

Boron injection is...

A. NOT required.

B. required based on Torus water temperature, but NOT due to the magnitude of Reactor power oscillations.

C. required based on the magnitude of Reactor power oscillations, but NOT due to Torus water temperature.

D. required based on both the magnitude of Reactor power oscillations and Torus water temperature.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	T-101 step RC/Q-9 requires boron injection before Torus water temperature reaches the threshold of 110°F. Since Torus water temperature is 112°F, boron injection is required. T-101 step RC/Q-8 requires boron injection if Reactor power oscillations exceed 25% peak to peak. Since oscillations are only 10% peak to peak, this step does not require boron injection.
Distracters:	A	T-101 step RC/Q-9 requires boron injection before Torus water temperature reaches the threshold of 110°F. Since Torus water temperature is 112°F, boron injection is required. Plausible because Torus water temperature is below other EOP limits (120°F, 130°F, HCTL).
	C	T-101 step RC/Q-8 requires boron injection if Reactor power oscillations exceed 25% peak to peak. Since oscillations are only 10% peak to peak, this step does not require boron injection. Plausible because power is both above APRM downscales and oscillating significantly. T-101 step RC/Q-9 requires boron injection before Torus water temperature reaches the threshold of 110°F. Since Torus water temperature is 112°F, boron injection is required. Plausible because Torus water temperature is below other EOP limits (120°F, 130°F, HCTL).
	D	T-101 step RC/Q-8 requires boron injection if Reactor power oscillations exceed 25% peak to peak. Since oscillations are only 10% peak to peak, this step does not require boron injection. Plausible because power is both above APRM downscales and oscillating significantly.

Question 52 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659123
User-Defined ID:	
Cross Reference Number:	
Topic:	Boron injection based on Torus temp / power oscillations
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(10)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-101	
Learning Objective:	PLOT-PBIG-2101 6	
K/A System:	295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	Importance; RO 4.0
K/A Statement:	A2.04 - Ability to determine and/or interpret the following as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Suppression pool temperature	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

53 **ID: 1649431** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- An electrical loss results in multiple alarms and indications.
- The following RCIC components have NEITHER green NOR red lights lit on Panel 20C004C:
  - MO-131, Supply
  - MO-21, To Feed Line

Which one of the following identifies the electrical power supply loss that results in these indications?

A. Division I DC

B. Division II DC

C. Division I Vital AC

D. Division II Vital AC

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	Division I DC power (2DA) supplies power to these RCIC components. Loss of this supply causes loss of both red and green indicating lights.
Distracters:	B	Division I DC power (2DA) supplies power to these RCIC components. Plausible because Division II DC power supplies HPCI and loss would give similar indications.
	C	Division I DC power (2DA) supplies power to these RCIC components. Plausible because Division I Vital AC power supplies many important components and loss would give similar indications.
	D	Division I DC power (2DA) supplies power to these RCIC components. Plausible because Division II Vital AC power supplies many important components and loss would give similar indications.

Question 53 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649431
User-Defined ID:	
Cross Reference Number:	
Topic:	Loss of indication on MO-131 and MO-21
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (8)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank **(2015 SYSID)** Bank ILT Exam Bank	Other Exam
Reference(s):	COL 13.1.A-2, E-26	
Learning Objective:	PLOT-5013 2b	
K/A System:	295004 Partial or Total Loss of DC Pwr	Importance; RO 3.2
K/A Statement:	A2.04 - Ability to determine and/or interpret the following as they apply to Partial or Total Loss of DC Pwr: System lineups	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

54 **ID: 1649435** **Points: 1.00**

Both Units are operating at 100% power with the following:

- The 4KV system is in a normal line-up, except for the E-312 breaker, which is racked out for elevator mechanism preventative maintenance.
- The control switch for the E-212 breaker is inadvertently bumped to the green-flag position and the breaker opens.

Which one of the following describes the response of the E-1 EDG and the impact on Technical Specifications?

Based on these conditions, the E-1 EDG __(1)__ automatically start. Entry into Technical Specifications is __(2)__.

A. (1) will
(2) required

B. (1) will
(2) NOT required

C. (1) will NOT
(2) required

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

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	The E-212 and the E-312 breakers are the normal and alternate feeder breakers for the E-12 bus. With E-312 breaker racked out and E-212 breaker taken to OPEN (both breakers "green-flagged"), EDG auto-start is defeated. Tech Spec (3.8.7) entry is required since the E-12 4kV bus is considered inoperable when de-energized.
Distracters:	A	The EDG will not autostart. Plausible that EDG would still autostart on undervoltage and just not close in on the bus.
	B	The EDG will not autostart. Plausible that EDG would still autostart on undervoltage and just not close in on the bus. Tech Spec entry is required. Plausible for an RO candidate that a single 4 KV bus being de-energized would not require Tech Spec entry due to designed redundancy.
	D	Tech Spec entry is required. Plausible for an RO candidate that a single 4 KV bus being de-energized would not require Tech Spec entry due to designed redundancy.

Question 54 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649435
User-Defined ID:	
Cross Reference Number:	
Topic:	Loss of E-12, EDG response and TS entry
Num Field 1:	
Num Field 2:	
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(8)

Source Documentation

Source:	New Exam item **(2008 NRC #19)** **X** Previous NRC Exam Modified Bank Other Exam Bank **X** ILT Exam Bank	
Reference(s):	PLOT-5054; SO 52A.1.B; Tech Spec 3.8.7	
Learning Objective:	PLOT-5054 5a	
K/A System:	295003 Partial or Complete Loss of AC	Importance; RO 3.1
K/A Statement:	2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

55 **ID: 1649437** **Points: 1.00**

An Alert was just declared on Unit 2 for a leak in the Reactor Coolant System that resulted in high Drywell pressure.

Which one of the following describes the required notification of the Nuclear Regulatory Commission (NRC) as the NRC Communicator in accordance with EP-AA-114, Notifications?

A. Place a Emergency Notification System (ENS) telephone call as soon as possible, but within a maximum of 15 minutes.

B. Place a Emergency Notification System (ENS) telephone call as soon as possible, but within a maximum of 1 hour.

C. Fax the Emergency Notification Worksheet as soon as possible, but within a maximum of 15 minutes.

D. Fax the Emergency Notification Worksheet as soon as possible, but within a maximum of 1 hour.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	The NRC is notified using the a Emergency Notification System (ENS) phone call with one hour per EP-AA-114.
Distracters:	A	The NRC is notified using the a Emergency Notification System (ENS) phone call with one hour per EP-AA-114. Plausible if the candidate confuses the NRC notification wit the state and local notification which is required within 15 minutes.
	C	The Emergency Notification worksheet is used to make the phone call to the NRC. There is not a requirement to fax the worksheet to the NRC. Plausible if the candidate confuses the ENW with the requirement to fax the state and local notification form..
	D	The Emergency Notification worksheet is used to make the phone call to the NRC. There is not a requirement to fax the worksheet to the NRC. Plausible if the candidate confuses the ENW with the requirement to fax the state and local notification form..

Question 55 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649437
User-Defined ID:	
Cross Reference Number:	2.4.30
Topic:	ILT-G5-7-003
Num Field 1:	2015 NRC
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(10)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank **(2015 NRC #72)** SYSID Other Exam Bank ILT Exam Bank	
Reference(s):	EP-AA-114	
Learning Objective:	G5 7	
K/A System:	295024 High Drywell Pressure	Importance; RO 2.7
K/A Statement:	2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

56 **ID: 1649438** **Points: 1.00**

Unit 2 is shutdown with the following:

- Movement of ***recently irradiated*** fuel assemblies is in progress.

The fuel pool gates are removed.

- Operations with the Potential to Drain the Reactor Vessel (OPDRVs) are in progress.

Then it is discovered that both trains of Standby Gas Treatment are inoperable.

Which one of the following describes the Technical Specification impact?

A. The fuel movement must be suspended immediately.
Action must also be initiated immediately to suspend the OPDRVs.

B. The fuel movement may continue for a maximum of one hour.
Action must be initiated immediately to suspend the OPDRVs.

C. The fuel movement must be suspended immediately.
Action must be initiated within a maximum of one hour to suspend the OPDRVs.

D. Both the fuel movement and OPDRVs may continue for a maximum of one hour.

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	TS 3.6.4.3 Condition E applies with 2 inoperable SBGT trains plus movement of recently irradiated fuel and OPDRVs in progress. TS 3.6.4.3 requires immediately suspending the fuel movement and immediately initiating action to suspend OPDRVs.
Distracters:	B	TS 3.6.4.3 requires immediately suspending the fuel movement. Plausible because many similar TSs allow 1 hour.
	C	TS 3.6.4.3 requires immediately initiating action to suspend OPDRVs. Plausible because many similar TSs allow 1 hour.
	D	TS 3.6.4.3 requires immediately suspending the fuel movement and immediately initiating action to suspend OPDRVs. Plausible because many similar TSs allow 1 hour.

Question 56 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649438
User-Defined ID:	
Cross Reference Number:	
Topic:	TS 3.6.4.3 actions for no operable SBGT trains
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(10)

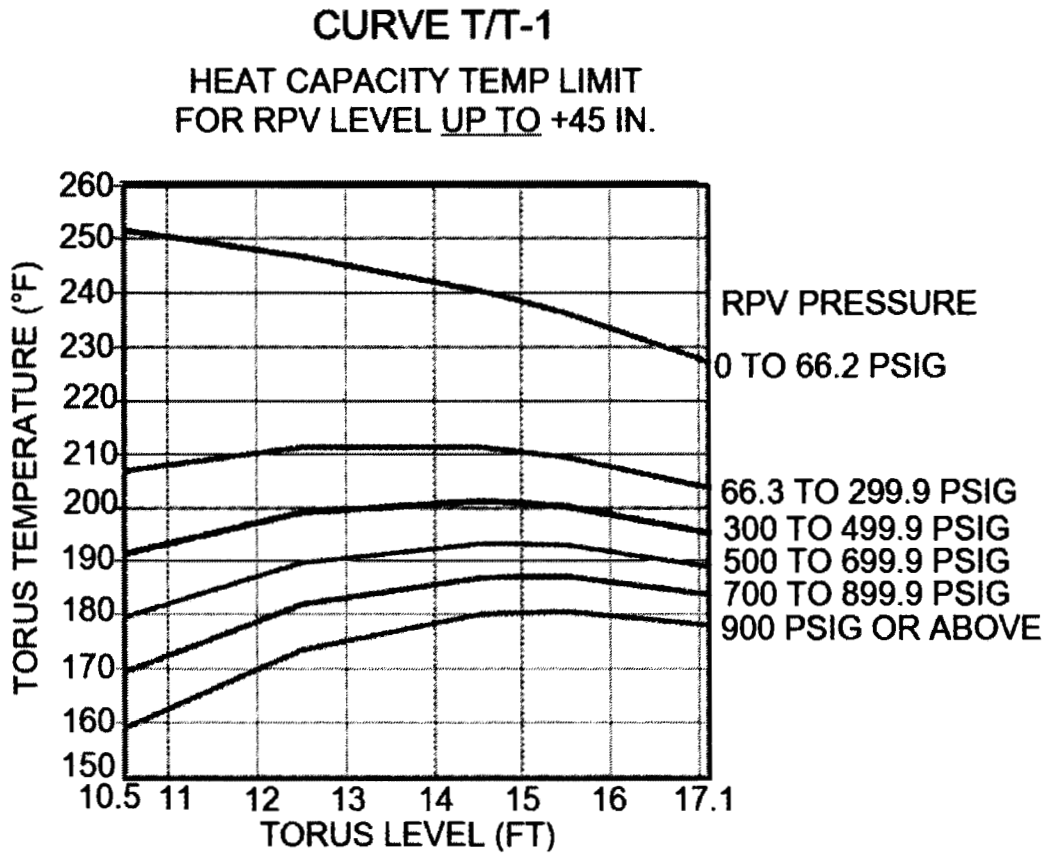
Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	TS 3.6.4.3	
Learning Objective:	PLOT-5009A 13	
K/A System:	295023 Refueling Accidents	Importance; RO 3.9
K/A Statement:	2.2.39 - Knowledge of less than or equal to one hour Technical Specification action statements for systems.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

57 **ID: 1659150** **Points: 1.00**

Unit 2 is operating at 100% power when a Main Turbine trip and high-power ATWS result in the following:

Parameter	Time 1	Time 2
Reactor water level	-185"	-175"
Reactor pressure	675 psig	825 psig
Torus water temperature	185°F	190°F
Torus water level	15.0'	15.1'



Which one of the following describes the operating point on the Heat Capacity Temperature Limit at Times 1 and 2, in accordance with T-102, Primary Containment Control?

A. SAT at both times

B. SAT at Time 1 and UNSAT at Time 2

C. UNSAT at Time 1 and SAT at Time 2

D. UNSAT at both times

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	At Time 1 with Torus temperature at 185°F and Torus level at 15.0', operation is below the applicable 500-699.9 psig HCTL curve, which is the SAT side of the curve. At Time 2 with Torus temperature at 190°F and Torus level at 15.1', operation is above the applicable 700-899.9 psig HCTL curve, which is the UNSAT side of the curve.
Distracters:	A	Conditions are UNSAT at Time 2. Plausible if candidate mis-plots point, uses incorrect pressure curve, or misunderstands which side of curve is SAT/UNSAT.
	C	Conditions are SAT at Time 1 and UNSAT at Time 2. Plausible if candidate mis-plots point, uses incorrect pressure curve, or misunderstands which side of curve is SAT/UNSAT.
	D	Conditions are SAT at Time 1. Plausible if candidate mis-plots point, uses incorrect pressure curve, or misunderstands which side of curve is SAT/UNSAT.

Question 57 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659150
User-Defined ID:	
Cross Reference Number:	
Topic:	HCTL interpretation
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(10)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-102	
Learning Objective:	PLOT-PBIG-2102 5	
K/A System:	295026 Suppression Pool High Water Temp.	Importance; RO 4.1
K/A Statement:	A2.01 - Ability to determine and/or interpret the following as they apply to Suppression Pool High Water Temp.: Suppression pool water temperature	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

58 **ID: 1649470** **Points: 1.00**

Both Units are operating at 100% power when Annunciator 007 J-6b, RADWASTE BLDG. STANDBY GAS TREAT. HEAT DET/FLOW FILTER A ELEV. 91'-6", alarms.

Which one of the following describes the associated fire-fighting system and how it is initiated?

A. Manually initiated Cardox

B. Automatically initiated Cardox

C. Manually initiated Water Deluge

D. Automatically initiated Water Deluge

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	This alarm is received based on detectors in the SBGT train A filter train. These detectors initiate a control room alarm, but do not automatically reposition any valves to initiate fire suppression. The associated fire suppression system is a water deluge system that must be initiated manually by opening valves.
Distracters:	A	The associated fire suppression system is a water deluge system that must be initiated manually. Plausible because adding water to a charcoal filter is generally undesirable, such that use of Cardox for fire suppression would be a possible alternative.
	B	The associated fire suppression system is a water deluge system that must be initiated manually. Plausible because adding water to a charcoal filter is generally undesirable, such that use of Cardox for fire suppression would be a possible alternative. Plausible that the same detectors that initiate the control room alarm would automatically initiate suppression, as in many other systems.
	D	The associated fire suppression system is a water deluge system that must be initiated manually. Plausible that the same detectors that initiate the control room alarm would automatically initiate suppression, as in many other systems.

Question 58 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649470
User-Defined ID:	
Cross Reference Number:	
Topic:	Fire suppression / heat detection alarm
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(4)

Source Documentation

Source:	New Exam item **(2007 NRC #20)** Modified Bank Bank **X** ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	ARC 007 J-6b	
Learning Objective:	PLOT-5009A 8c	
K/A System:	600000 Plant Fire On Site	Importance; RO 2.6
K/A Statement:	K2.01 - Knowledge of the interrelations between Plant Fire On Site and the following: Sensors, detectors and valves	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

59 **ID: 1649471** **Points: 1.00**

Unit 2 is operating at 97% power following an inadvertent reactivity addition with the following portion of a P1 case:

MOST LIMITING LOCATIONS (NON-SYMMETRIC)

MFLCPR	LOC	MFLPD	LOC	MAPRAT	LOC
0.918	29-46	1.078	9-36- 5	0.835	9-36- 5

Which one of the following describes the status of thermal limits?

A. All thermal limits are SAT.

B. The Linear Heat Generation Rate (LHGR) thermal limit is UNSAT.

C. The Minimum Critical Power Ratio (MCPR) thermal limit is UNSAT.

D. The Average Planar Linear Heat Generation Rate (APLHGR) thermal limit is UNSAT.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	With MFLPD greater than 1.0, the associated Linear Heat Generation Rate (LHGR) thermal limit is UNSAT. **Note:** The question meets the K/A because it presents a situation where thermal limits are being affected by an inadvertent reactivity addition and tests the operational implications of thermal limits (how to interpret a P1 case to determine that status of thermal limits).
Distracters:	A	With MFLPD greater than 1.0, the associated Linear Heat Generation Rate (LHGR) thermal limit is UNSAT. Plausible if the candidate does not know how to read a P1 case or understand each of the thermal limit ratios must be less than or equal to 1.0.
	C	With MFLCPR less than 1.0, the associated Minimum Critical Power Ratio (MCPR) thermal limit is SAT. Plausible if the candidate does not know how to read a P1 case, understand each of the thermal limit ratios must be less than or equal to 1.0, or understand which ratio corresponds to which thermal limit.
	D	With MAPRAT less than 1.0, the associated Average Planar Linear Heat Generation Rate (APLHGR) thermal limit is SAT. Plausible if the candidate does not know how to read a P1 case, understand each of the thermal limit ratios must be less than or equal to 1.0, or understand which ratio corresponds to which thermal limit.

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649471
User-Defined ID:	
Cross Reference Number:	
Topic:	MFLPD violation
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ST-O-098-01D-2	
Learning Objective:	PLOT-1870 31	
K/A System:	295014 Inadvertent Reactivity Addition	Importance; RO 3.7
K/A Statement:	K1.05 - Knowledge of the operational implications of the following concepts as they apply to Inadvertent Reactivity Addition: Fuel thermal limits	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

60 **ID: 1649472** **Points: 1.00**

Unit 3 is operating at 90% power with the following:

- The A pressure regulator is in control.
- The B pressure regulator is set as the backup pressure regulator.

Then, the Pressure Averaging Manifold (PAM) pressure signal to the A pressure regulator drifts downscale.

Which one of the following describes the Reactor pressure response?

Reactor pressure...

A. rises and causes a Reactor scram.

B. lowers and causes a Reactor scram.

C. rises and stabilizes at a higher value without causing a Reactor scram.

D. lowers and stabilizes at a lower value without causing a Reactor scram.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	With the A pressure regulator initially in control, pressure setpoint bias is set such that the B pressure regulator will control Reactor pressure approximately 3 psig higher than the A pressure regulator. When the PAM pressure signal fails low to the A pressure regulator, it begins to close TCVs to stop the sensed drop in pressure. This causes actual Reactor pressure and PAM pressure to rise. As the B pressure regulator senses this rise in pressure, its input summer error signal grows and overcomes the error signal from the A pressure regulator. EHC logic is setup so that the pressure regulator with the higher error signal controls pressure. Therefore, the B pressure regulator comes into control and stabilizes Reactor pressure approximately 3 psig higher than the initial value. This prevents a Reactor scram on high pressure (nominally at 1085 psig), which would otherwise occur.
Distracters:	A	The B pressure regulator limits the Reactor pressure rise to approximately 3 psig. Plausible because on the opposite failure, the Reactor pressure change would cause a scram on MSIV closure.
	B	Reactor pressure rises, not lowers. Plausible because the opposite failure would cause Reactor pressure to lower. The B pressure regulator limits the Reactor pressure change to approximately 3 psig. Plausible because on the opposite failure, the Reactor pressure change would cause a scram on MSIV closure.
	D	Reactor pressure rises, not lowers. Plausible because the opposite failure would cause Reactor pressure to lower.

Question 60 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649472
User-Defined ID:	
Cross Reference Number:	
Topic:	PAM pressure signal fails low
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	SO 1B.1.A-3, OT-102	
Learning Objective:	PLOT-5001DL 6b	
K/A System:	295007 High Reactor Pressure	Importance; RO 3.5
K/A Statement:	K2.01 - Knowledge of the interrelations between High Reactor Pressure and the following: Reactor/turbine pressure regulating system	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

61 **ID: 1649474** **Points: 1.00**

Which one of the following describes the reason for performing a Reactor scram due to high Secondary Containment temperatures in accordance with T-103, Secondary Containment Control, bases?

A. Reduces the energy that is being discharged into the Secondary Containment to decay heat levels.

B. Continued Reactor operation is NOT allowed once all Reactor water level indications become inoperable.

C. Continued Reactor operation is NOT allowed once the Secondary Containment design temperature is exceeded.

D. Allows reducing the heat load in the Secondary Containment by securing systems only needed with the Reactor operating at power.

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	T-103 bases state, "Direction is provided to perform a GP-4, "Manual Reactor Scram" in order to reduce the energy that is being discharged into the Secondary Containment to decay heat levels."
Distracters:	B	The specific basis is to reduce the energy that is being discharged into the Secondary Containment to decay heat levels. Plausible because elevated Reactor Building temperatures do affect Reactor water level indications and this is discussed in T-103.
	C	The specific basis is to reduce the energy that is being discharged into the Secondary Containment to decay heat levels. Plausible because high Secondary Containment temperatures do challenge the various Reactor Building design temperatures. However, the values used in T-103 to initiate a Reactor scram are not directly tied to building design temperatures (they are generally above design temperature).
	D	The specific basis is to reduce the energy that is being discharged into the Secondary Containment to decay heat levels. Plausible because another method to attempt to reduce Secondary Containment temperatures would be to secure operating equipment in the building, some of which should not be secured with the Reactor at power.

EXAMINATION ANSWER KEY

2017 RO NRC Exam

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649474
User-Defined ID:	
Cross Reference Number:	
Topic:	Basis for scram on high SC temp
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-103 bases	
Learning Objective:	PLOT-PBIG-2103 6	
K/A System:	295032 High Secondary Containment Area Temperature	Importance; RO 3.6
K/A Statement:	K3.02 - Knowledge of the reasons for the following responses as they apply to High Secondary Containment Area Temperature: Reactor SCRAM	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

62 **ID: 1649475** **Points: 1.00**

Unit 2 is operating at 90% power with the following:

- The 2A Reactor Feedwater pump trips.
- Reactor water level drops to +15" before turning and beginning to rise.

Which one of the following describes the resulting status of the Reactor Recirculation pumps?

A. Running at original speed

B. Run back to 45% speed

C. Run back to 30% speed

D. Tripped

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	The combination of an individual Reactor Feedwater pump flow below 20% and Reactor water level lowering below +17" causes Recirc pumps to run back to 45% speed.
Distracters:	A	Recirc pumps automatically runback to 45% speed. Plausible because if level did not lower below +17", the Feedwater pump trip alone would not have cause a run back.
	C	Recirc pumps automatically runback to 45% speed. Plausible because a 30% run back exists for other conditions, including Reactor water level <+17" with a scram present.
	D	Recirc pumps continue to operate. Plausible because a Recirc pump trip exists for a lower Reactor water level (-48").

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649475
User-Defined ID:	
Cross Reference Number:	
Topic:	45% runback on FWP trip and +15"
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(3)

Source Documentation

Source:	New Exam item Previous NRC Exam **X** Modified Bank **(SYSID 994759)** Other Exam Bank ILT Exam Bank	
Reference(s):	OT-100	
Learning Objective:	PLOT-5002 5v	
K/A System:	295009 Low Reactor Water Level	Importance; RO 3.0
K/A Statement:	A1.03 - Ability to operate and/or monitor the following as they apply to Low Reactor Water Level: Recirculation system: Plant-Specific	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

63 **ID: 1649499** **Points: 1.00**

Unit 2 has experienced a loss of Drywell cooling and a steam leak in the Drywell with the following:

- Drywell spray has been initiated due to high Drywell temperature per T-102, Primary Containment Control.
- Drywell temperature is 230°F and lowering slowly.
- Drywell pressure is 6 psig and lowering slowly.

Which one of the following identifies when Drywell spray is required to be secured in accordance with T-102?

Secure Drywell spray before Drywell...

A. pressure drops to the threshold value of 2 psig.

B. pressure drops to the threshold value of 0 psig.

C. temperature drops to the threshold value of 200°F.

D. temperature drops to the threshold value of 145°F.

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	T-102 requires securing Drywell spray based on Drywell pressure, not temperature. The threshold value for securing Drywell spray is 2 psig in T-102. **Note:** The question meets the K/A by presenting a situation where Drywell sprays have been initiated due to high Drywell temperature and requiring knowledge of when Drywell sprays are terminated based on Drywell pressure.
Distracters:	B	The threshold value for securing Drywell spray is 2 psig in T-102, not 0 psig. Plausible because 0 psig would prevent negative pressure in the Drywell and is the limit at some other plants in the SAMPs.
	C	T-102 requires securing Drywell spray based on Drywell pressure, not temperature. Plausible because spray was initiated based on temperature and 200°F is a value used in the DW/T leg of T-102.
	D	T-102 requires securing Drywell spray based on Drywell pressure, not temperature. Plausible because spray was initiated based on temperature and 145°F is a value used in the DW/T leg of T-102.

Question 63 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649499
User-Defined ID:	
Cross Reference Number:	
Topic:	When to terminate sprays
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
LOW			10CFR55.41(b) (10)

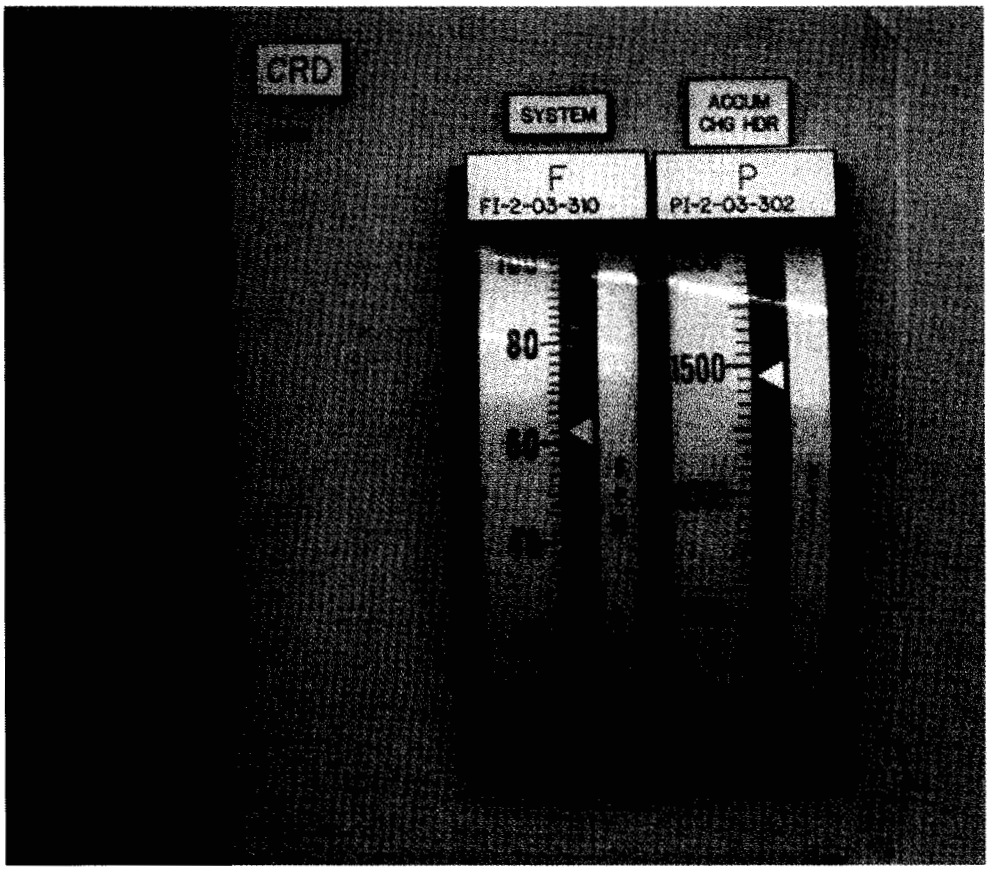
Source Documentation		
Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-102	
Learning Objective:	PLOT-PBIG-2102 5	
K/A System:	295012 High Drywell Temperature	Importance; RO 3.9
K/A Statement:	A2.02 - Ability to determine and/or interpret the following as they apply to High Drywell Temperature: Drywell pressure	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

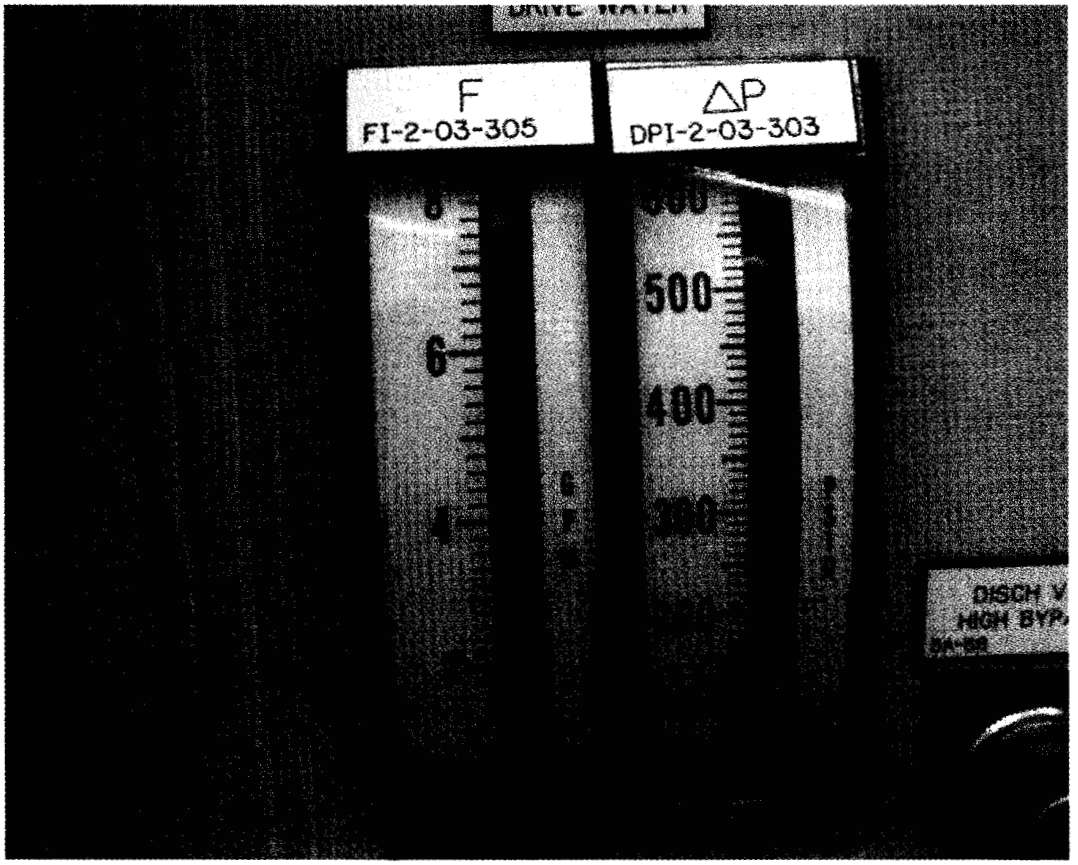
64 **ID: 1659027** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- The A CRD pump has tripped.
- ON-107, Loss of CRD Regulating Function, is being executed.
- The B CRD pump has been started.

Which statement is true given the above conditions?





A. Throttle closed MO-2-3-20 "Drive Water Pressure" to establish proper system differential pressure.

B. Throttle open MO-2-3-20 "Drive Water Pressure" to establish proper system differential pressure.

C. Lower the setpoint on the CRD flow valve controller to establish proper system flow.

D. Raise the setpoint on the CRD flow valve controller to establish proper system flow.

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	System differential pressure is required to be 260 to 280 psid as read on DPI-303. With differential pressure less than this value, MO-20 must be closed to raise drive water pressure. System flow is in the proper band of 55 to 65 gpm.
Distracters:	B	System differential pressure is low. Plausible if the candidate doesn't realize the MO-20 must be closed to adjust system differential pressure and not in the open direction.
	C	System flow is in the proper band of 55 to 65 gpm. Plausible if the candidate does not recall the proper system flow rate.
	D	System flow is in the proper band of 55 to 65 gpm. Plausible if the candidate does not recall the proper system flow rate.

EXAMINATION ANSWER KEY

2017 RO NRC Exam

Question 64 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659027
User-Defined ID:	
Cross Reference Number:	
Topic:	ILT-5003B-002
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(6)

Source Documentation

Source:	**X** New Exam item Exam Modified Bank Bank ILT Exam Bank	Previous NRC Other Exam
Reference(s):	ON-102, SO 3.1.B-2	
Learning Objective:	PLOT-5003A 9g	
K/A System:	295022 Loss of CRD Pumps	Importance; RO 4.6
K/A Statement:	2.1.31 - Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

65 **ID: 1649503** **Points: 1.00**

A Unit 2 startup is in progress when a loss of coolant accident results in the following:

- RPV level dropped to -220 inches
- Reactor water level is -10" and rising.
- Reactor pressure is 75 psig and slowly lowering.
- Condensate, Core Spray and RHR are injecting to the Reactor.
- Torus water level is 17.1' and slowly rising.
- It has been determined that Reactor injection sources must be controlled to maintain Torus water level below the SRV Tail Pipe Limit.

Which one of the following describes the required control of the available injection sources in accordance with T-101, RPV Control, and T-102, Primary Containment Control?

Injection with...

A. Condensate, Core Spray, and RHR are all equally acceptable.

B. Condensate should be minimized.

C. Core Spray should be minimized.

D. RHR should be minimized.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	With the SRV tailpipe level limit challenged, T-102 step T/L-26 states, "If ACC can be assured, then terminate injection into the RPV from sources external to Containment except for system required to shutdown the Reactor." Adequate core cooling is assured with Reactor water level near the normal band and Core Spray also available to inject. Condensate injects water from external to the Containment and therefore should be terminated.
Distracters:	A	Condensate injects water from external to the Containment and therefore should be terminated. Plausible because Condensate injection is still allowed if needed for adequate core cooling.
	C	Core Spray injects water from internal to the Containment and therefore is not restricted. Plausible because injection sources must be restricted and in some situations Core Spray injection is restricted (ATWS).
	D	RHR injects water from internal to the Containment and therefore is not restricted. Plausible because injection sources must be restricted and in some situations RHR injection is restricted (initial level reduction during ATWS).

Question 65 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649503
User-Defined ID:	
Cross Reference Number:	
Topic:	High Torus level - Control of injection sources
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank **(JAF 9/14 NRC #59)** Other Exam Bank ILT Exam Bank	
Reference(s):	T-102	
Learning Objective:	PLOT-PBIG-2102 5	
K/A System:	295029 High Suppression Pool Wtr Lvl	Importance; RO 3.8
K/A Statement:	2.4.9 - Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

66 **ID: 1649507** **Points: 1.00**

Given the following individuals:

(1) A qualified, non-licensed Operator who has been selected for license class but has not yet begun formal license class training.

(2) An instant SRO license candidate who is performing the pre-license class plant familiarization guide.

(3) An RO license candidate who is completing the in-plant OJT phase of license class.

Which one of the following identifies the individual(s) that may be allowed to perform reactivity manipulations in the plant under the supervision of a licensed Reactor Operator, in accordance with OP-AA-300, Reactivity Management?

A. (3) only

B. (1) and (3) only

C. (2) and (3) only

D. (1), (2), and (3)

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	OP-AA-300 allows a licensed RO to supervise reactivity manipulations in the plant by non-licensed individuals only if the non-licensed individual is enrolled in an approved training program. TQ-AA-150 is the associated training program and it restricts such reactivity manipulations to individuals in the OJT phase of the ILT program that have completed prerequisite training (GFEs and systems training) and have specific approval to then perform reactivity manipulations. Only Individual (3) meets all these requirements.
Distracters:	B	Individual (1) is not allowed to perform reactivity manipulations under the guidance of a licensed RO. Plausible because Individual (1) is qualified to perform non-licensed activities and will eventually perform reactivity manipulations as a license candidate, however since they are not yet in the formal training class, they are not yet allowed to perform reactivity manipulations under instruction.
	C	Individual (2) is not allowed to perform reactivity manipulations under the guidance of a licensed RO. Plausible because Individual (2) is in a preparatory phase of license class, but has not yet completed the prerequisite training for reactivity manipulations.
	D	Individual (1) is not allowed to perform reactivity manipulations under the guidance of a licensed RO. Plausible because Individual (1) is qualified to perform non-licensed activities and will eventually perform reactivity manipulations as a license candidate, however since they are not yet in the formal training class, they are not yet allowed to perform reactivity manipulations under instruction. Individual (2) is not allowed to perform reactivity manipulations under the guidance of a licensed RO. Plausible because Individual (2) is in a preparatory phase of license class, but has not yet completed the prerequisite training for reactivity manipulations.

Question 66 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649507
User-Defined ID:	
Cross Reference Number:	
Topic:	Supervision of non-licensed personnel performing reactivity manipulations
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
LOW			10CFR55.41(b) (10)

Source Documentation

Source:	New Exam item Modified Bank Bank **(NMP2 2014 Cert #74)** ILT Exam Bank	Previous NRC Exam **X** Other Exam
Reference(s):	OP-AA-300, TQ-AA-150	
Learning Objective:	PLOT-DBIG-1536 3	
K/A System:		Importance; RO 2.9
K/A Statement:	2.1.9 - Ability to direct personnel activities inside the control room.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

67 **ID: 1649619** **Points: 1.00**

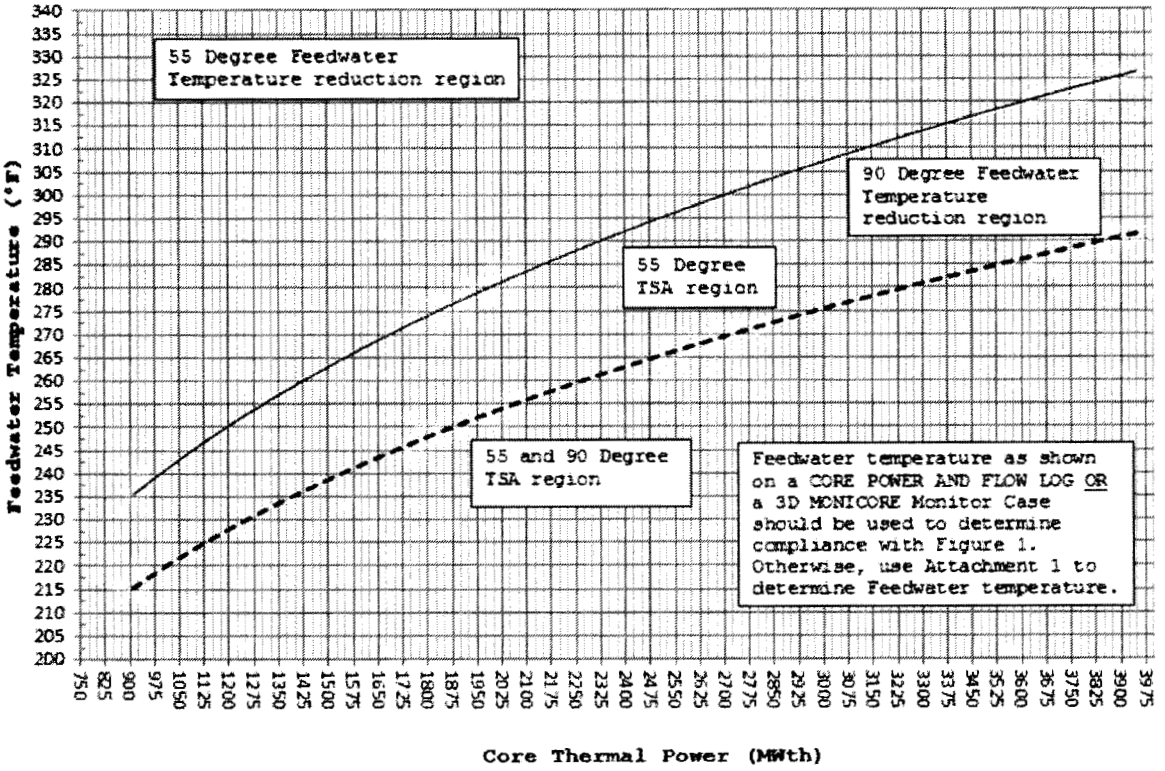
Unit 2 has experienced a loss of Feedwater heating with the following:

- Reactor power is 73%
- Feedwater temperature is 277°F.

Which one of the following describes the operating point on OT-104, Positive Reactivity Insertion, Figure 1, Feedwater Temperature Limits (given below)?

FIGURE 1

FEEDWATER TEMPERATURE LIMITS



(1) If the 55°F temperature reduction limit must be used, then the operating point is...

(2) If the 90°F temperature reduction limit is allowed, then the operating point is...

A. (1) SAT
(2) SAT

B. (1) SAT

(2) UNSAT

C. (1) UNSAT
(2) SAT

D. (1) UNSAT
(2) UNSAT

Answer: C

Answer Explanation		
Choice		Basis or Justification
Correct:	C	Core thermal power is approximately 2884 MWth (0.73 x 3951 MWth). Plotting a Feedwater temperature of 277°F with this power level results in operation between the two curves. This region is SAT for the 90°F limit, but UNSAT for the 55°F limit ("TSA region" below each respective curve is UNSAT for that limit and requires power reduction).
Distracters:	A	The operating point is UNSAT for the 55°F limit. Plausible if the candidate makes an error in plotting the data or does not understand the allowable regions of the limit.
	B	The operating point is UNSAT for the 55°F limit and SAT for the 90°F limit. Plausible if the candidate makes an error in plotting the data or does not understand the allowable regions of the limit.
	D	The operating point is SAT for the 90°F limit. Plausible if the candidate makes an error in plotting the data or does not understand the allowable regions of the limit.

Question 67 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649619
User-Defined ID:	
Cross Reference Number:	
Topic:	Feedwater temperature graph
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	OT-104	
Learning Objective:	PLOT-DBIG-1540 7	
K/A System:		Importance; RO 4.1
K/A Statement:	2.1.43 - Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

68 **ID: 1649621** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- An inspection must be performed on a 120 VAC electrical panel.
- The Electricians performing the inspection need to be able to open the panel's disconnect switch for personnel protection during some portions of the inspection.
- The Electricians performing the inspection also need to be able to close the panel's disconnect switch for periodic verifications.
- It is desired for the Tagout to be continuously hung during the activity, so repeated tag clearing and re-hanging is NOT required.

Which one of the following describes an allowable tagging arrangement for this maintenance activity in accordance with OP-MA-109-101, Clearance and Tagging?

Tag the electrical panel's disconnect switch with...

A. a Danger Tag only.

B. an Information Tag only.

C. a Special Condition Tag only.

D. both a Danger Tag and a Special Condition Tag simultaneously.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	A Special Condition Tag has equal authority to a Danger Tag to provide personnel protection, but also allows the tagged component to be manipulated without clearing the tag, as required by this maintenance activity.
Distracters:	A	A Danger Tag would not allow re-positioning the disconnect switch without clearing the tag. Plausible because the Danger Tag does provide the personnel protection required.
	B	An Information Tag does not provide the required personnel protection. Plausible because the Information Tag would allow re-positioning the disconnect switch without clearing the tag.
	D	A Danger Tag and Special Condition Tag are not allowed to be placed on the same component at the same time. Plausible because the Danger Tag does provide the personnel protection required and the Special Condition Tag is used to allow manipulation of a tagged component.

Question 68 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649621
User-Defined ID:	
Cross Reference Number:	
Topic:	SCT
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item Modified Bank Bank **(NMP1 2013 NRC #71)** ILT Exam Bank	Previous NRC Exam **X** Other Exam
Reference(s):	OP-MA-109-101	
Learning Objective:	O2.0-007	
K/A System:		Importance; RO 2.6
K/A Statement:	2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

69 **ID: 1649514** **Points: 1.00**

A Unit 2 startup is in progress in accordance with GP-2-2, Normal Plant Start-up. Control rods are being withdrawn to achieve criticality.

Which one of the following describes:

(1) the WRNM count rate at which continuous control rod withdrawal ***first*** becomes restricted,

and

(2) the associated restriction, in accordance with GP-2-2?

Note: Assume NO other specific direction has been given by Reactor Engineering.

A. (1) Two doublings
(2) Notch withdrawal required at all positions from 00 to 48.

B. (1) Two doublings
(2) Notch withdrawal required only at positions from 04 to 36.

C. (1) Three doublings
(2) Notch withdrawal required from position 00 to position 48.

D. (1) Three doublings
(2) Notch withdrawal required only at positions from 04 to 36.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	GP-2 cautions require, prior to criticality, when the count rate on WRNM exceeds three doublings, all rods be notch withdrawn from position 04 to 36. Continuous withdrawal is specifically allowed from 00 to 04 to allow double clutching. Continuous withdrawal is specifically allowed from 36 to 48 due to low rod worth.
Distracters:	A	GP-2 restricts continuous withdrawal after three doublings, not two. Plausible because two doublings is a significant rise in power and near the correct answer. GP-2 only restricts continuous withdrawal from positions 04 to 36. Plausible because most rod positions are restricted and high worth could be found in other regions.
	B	GP-2 restricts continuous withdrawal after three doublings, not two. Plausible because two doublings is a significant rise in power and near the correct answer.
	C	GP-2 only restricts continuous withdrawal from positions 04 to 36. Plausible because most rod positions are restricted and high worth could be found in other regions.

Question 69 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649514
User-Defined ID:	
Cross Reference Number:	
Topic:	GP-2 restriction on continuous rod withdrawal
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
Memory			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item NRC Exam **(SSES LOC27 NRC #22)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	GP-2	
Learning Objective:	PLOT-DBIG-1536 4	
K/A System:		Importance; RO 4.1
K/A Statement:	2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

70 **ID: 994072** **Points: 1.00**

A transient on Unit 3 has resulted in the following:

- Significant fuel damage has occurred.
- The Reactor Building has become a High Radiation Area (General Area dose rates of 120 mR/hr) and has no current valid Radiation Work Permit (RWP).
- Operations personnel must enter the Reactor Building for one hour to help mitigate the transient.
- No dose extensions are required.

Which one of the following describes the ***minimum*** requirement for an operator to enter this area in accordance with RP-AA-403, Administration of the Radiation Work Permit Program?

The ***minimum*** requirement for an operator to enter the area is that they must have...

A. permission from the Radiation Protection Manager.

B. coverage by a qualified Radiation Protection Technician.

C. permission from the Emergency Director after Emergency Plan activation.

D. an extra Electronic Dosimeter with a dose alarm setpoint less than 500 mRem.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	RP-AA-403 section 4.7 provides guidance for entering an area without a valid RWP during emergency conditions. The procedure requires coverage by a qualified Radiation Protection Technician. **Note:** The question meets the K/A by presenting a radiation hazard that has arisen during emergency conditions (Reactor Building high radiation) and requiring knowledge of the requirement for entering the area with this radiation hazard still in place.
Distracters:	A	The procedure requires the RPT to notify RP Management as soon as possible, but their permission is not required prior to entry. Plausible because notifying RP supervision is required as soon as possible.
	C	The EDs permission is not required unless a dose extension is required for entry into the High Radiation Area. Plausible because ED permission would be required if an emergency dose extension were necessary.
	D	An electronic dosimeter is required, but not an additional one with a setpoint less than 500 mRem. Plausible because an additional electronic dosimeter does provide extra protection against malfunction, and a 500 mRem dose alarm setpoint is a threshold used in RP-AA-403 for extra risk management.

Question 70 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	1.00
System ID:	994072
User-Defined ID:	ILT-1760-4-002
Cross Reference Number:	295033 EK1.02
Topic:	ILT-1760-4-002 entry into high rad area
Num Field 1:	
Num Field 2:	A NRC
Text Field:	B

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
Memory			10CFR55.41(b) (12)

Source Documentation

Source:	New Exam item **(2009 NRC #70)** Modified Bank Bank **X** ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	RP-AA-403, RP-AA-460	
Learning Objective:	PLOT-1760 4	
K/A System:		Importance; RO / SRO 4.1/ 4.0
K/A Statement:	2.3.14 - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

71 **ID: 1649557** **Points: 1.00**

Which one of the following radiation monitors would directly require entry into T-103, Secondary Containment Control, given a valid upscale reading?

A. Area radiation monitor 1.3, Torus Compartment

B. Area radiation monitor 3.1, Heater & RFPT Area-South

C. Process radiation monitor RI-0-17-050A, Main Stack Exhaust

D. Process radiation monitor RIS-2-17-150A, Air Ejector Discharge

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	T-103 entry is required given any Reactor Building Area Radiation Level above the Action Level, as given in the associated table SC/R-1. ARM 1.3, Torus Compt being above 9 x 10^3 mr/hr requires T-103 entry.
Distracters:	B	ARM 3.1, Heater & RFPT Area-South, does not require T-103 entry because it indicates high radiation in the Turbine Building, not Reactor Building. Plausible because it is an ARM like the correct answer and could indicate primary system leakage that would affect T-104 implementation.
	C	The Main Stack Exhaust rad monitors do not require T-103 entry. Plausible because they relate to T-104 and EALs.
	D	The Air Ejector Discharge rad monitors do not require T-103 entry. Plausible because they relate to EALs and indicate fuel degradation.

Question 71 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649557
User-Defined ID:	
Cross Reference Number:	2.3.15
Topic:	Rad monitor requiring T-103 entry
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (11)

Source Documentation

Source:	**X** New Exam item NRC Exam **(JAF 9/14 NRC #70)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	T-103	
Learning Objective:	PLOT-PBIG-2103 1	
K/A System:		Importance; RO 2.9
K/A Statement:	2.3.15 - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

#72: QUESTION REDACTED DUE TO SECURITY-RELATED CONTENTS.

73 **ID: 1649623** **Points: 1.00**

Unit 2 has experienced a transient with the following:

- Reactor water level is +5" and rising slowly.
- Reactor pressure is 1095 psig and steady.
- Reactor power is downscale on APRMs.
- Drywell pressure is 1.7 psig and rising slowly.
- Drywell average temperature is 142°F and rising slowly.
- Torus water level is 15.2' and rising slowly.
- Torus temperature is 82°F and rising slowly.
- Annunciator 215 D-2, REAC BLDG EQUIPMENT DRAIN SUMP HI – HI LEVEL, is in alarm.

Which one of the following lists the Emergency Operating Procedures that are required to be entered based on current conditions?

A. T-101, RPV Control, and T-102, Primary Containment Control, only

B. T-101, RPV Control, and T-103, Secondary Containment Control, only

C. T-102, Primary Containment Control, and T-103, Secondary Containment Control, only

D. T-101, RPV Control, T-102, Primary Containment Control, and T-103, Secondary Containment Control

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	T-101 entry is required due to Reactor pressure above 1085 psig. T-102 entry is required due to Torus water level above 14.9'. T-103 entry is not required.
Distracters:	B	T-102 entry is required due to Torus water level above 14.9'. Plausible because multiple other conditions are elevated but below the T-102 entry condition. T-103 entry is not required. Plausible because the related Annunciator for hi-hi floor drain sump water level would require T-103 entry.
	C	T-101 entry is required due to Reactor pressure above 1085 psig. Plausible because Reactor water level is above the entry condition and Drywell pressure is below the entry condition. T-103 entry is not required. Plausible because the related Annunciator for hi-hi floor drain sump water level would require T-103 entry.
	D	T-103 entry is not required. Plausible because the related Annunciator for hi-hi floor drain sump water level would require T-103 entry.

Question 73 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649623
User-Defined ID:	
Cross Reference Number:	2.4.1
Topic:	Assess T-101, T-102, and T-103 entry requirements
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
HIGH			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item NRC Exam **(JAF 4/14 NRC #73)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	T-101, T-102, T-103	
Learning Objective:	PLOT-1560 1	
K/A System:		Importance; RO 4.6
K/A Statement:	2.4.1 - Knowledge of EOP entry conditions and immediate action steps.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

74 **ID: 1649568** **Points: 1.00**

An alarm is about to be received in the Main Control Room due to surveillance testing.

Given the following possible actions ***prior to*** receipt of the alarm:

(1) Discuss the alarm with the Control Room Supervisor.

(2) Visibly identify the alarm with a flagging tool.

(3) Review the associated Alarm Response Card.

Which one of the following identifies which of these actions is required prior to receipt of the alarm to consider it an "expected alarm" in accordance with OP-AA-103-102, Watchstanding Practices?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

D. (1), (2), and (3)

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	To call an alarm an "expected alarm", it must first be verified to be correct for plant conditions. OP-AA-103-102 contains the required actions for this verification. It is required to both review the Alarm Response Card and discuss the alarm with the Control Room Supervisor prior to receipt of the alarm. It is not required to visibly mark the alarm with a flagging tool. **Note:** The question meets the K/A by testing generic requirements for verifying an alarm is consistent with plant conditions during surveillance testing prior to receipt of the alarm to be in compliance with the "expected alarm" standards. If these prerequisites are not met, then other actions are required after receipt of the alarm to verify it is consistent with plant conditions.
Distracters:	A	It is also required to review the Alarm Response Card prior to receipt of the alarm. Plausible that only discussing with Control Room Supervisor would be required because the corrective actions in the ARC would not be necessary for a planned alarm during Surveillance Testing.
	B	It is not required to visibly mark the alarm with a flagging tool. Plausible because this is a common practice to bring attention to an off-normal alarm so that all shift personnel understand why it is in alarm. It is also required to review the Alarm Response Card prior to receipt of the alarm. Plausible that only discussing with Control Room Supervisor would be required because the corrective actions in the ARC would not be necessary for a planned alarm during Surveillance Testing.
	D	It is not required to visibly mark the alarm with a flagging tool. Plausible because this is a common practice to bring attention to an off-normal alarm so that all shift personnel understand why it is in alarm.

Question 74 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649568
User-Defined ID:	
Cross Reference Number:	2.4.46
Topic:	Expected alarm
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b) (10)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	OP-AA-103-102	
Learning Objective:	PLOT-DBIG-1529 1d	
K/A System:		Importance; RO 4.2
K/A Statement:	2.4.46 - Ability to verify that the alarms are consistent with the plant conditions.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

75 **ID: 1649572** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Annunciator 218 E-2, AIR EJECTOR DISCHARGE RADIATION HIGH, is in alarm.
- Air Ejector Discharge radiation has been verified to be above the alarm setpoint and slowly rising.

Which one of the following describes the required action in response to the high radiation level in accordance with the Alarm Response Cards?

A. Manually scram the Reactor now per GP-4.

B. Initiate a down power now per GP-9, but a scram is NOT required.

C. Manually scram the Reactor per GP-4 if Annunciator 218 E-1, AIR EJECTOR DISCHARGE RADIATION HIGH-HIGH, alarms.

D. Initiate a down power per GP-9 if Annunciator 218 E-1, AIR EJECTOR DISCHARGE RADIATION HIGH-HIGH, alarms, but a scram is NOT required.

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	A down power is required to limit the release from Offgas if Annunciator 218 E-1, AIR EJECTOR DISCHARGE RADIATION HIGH-HIGH, alarms.
Distracters:	A	A Reactor scram is not required. Plausible because high Air Ejector Discharge radiation causes a release and the release rate is significantly lowered if the Reactor is scrammed. Also plausible because other high radiation conditions require a scram (MSL).
	B	A down power is not required. Plausible because high Air Ejector Discharge radiation causes a release and the release rate is significantly lowered if Reactor power is lowered. Also plausible because if conditions worsen, a down power will be required.
	C	A Reactor scram is not required. Plausible because high Air Ejector Discharge radiation causes a release and the release rate is significantly lowered if the Reactor is scrammed. Also plausible because other high radiation conditions require a scram (MSL).

Question 75 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1649572
User-Defined ID:	
Cross Reference Number:	2.3.11
Topic:	Action for Air Ejector Discharge high rad
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	RO
MEMORY			10CFR55.41(b)(10)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC 218 E-2, ARC 218 E-1	
Learning Objective:	PLOT-5063 8a	
K/A System:		Importance; RO 3.8
K/A Statement:	2.3.11 - Ability to control radiation releases.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

EXAMINATION ANSWER KEY

2017 SRO NRC Exam

1 **ID: 1455828** **Points: 1.00**

Unit 2 has experienced a seismic event with the following:

- All control rods are fully inserted, except one control rod is stuck at position 48.
- Reactor pressure is 920 psig and stable on Turbine Bypass Valves.
- Torus water level is 9.9' and lowering.
- Operators have initiated all available Torus makeup.

Which one of the following describes the required control of Reactor pressure, in accordance with the Emergency Operating Procedures?

An Emergency Blowdown is...

A. required. Do NOT open SRVs.

B. required. SRVs are required to be opened.

C. NOT required. The cooldown rate may exceed 100°F/hr.

D. NOT required. The cooldown rate must stay below 100°F/hr.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	Since Torus water level is below 10.5', T-102 steps T/L-9 and 10 require an Emergency Blowdown using T-112. Since Torus water level is above 7' and Reactor pressure is more than 50 psig above Torus pressure, SRVs are required to be opened.
Distracters:	A	SRVs are required to be opened. Plausible because if Torus water level was below 7', then SRVs would not be opened.
	C	An Emergency Blowdown is required. Plausible because this would be the correct answer if Torus water level was above 10.5'.
	D	An Emergency Blowdown is required. Plausible because this would be the correct answer if Torus water level was above 10.5' and more control rods were withdrawn.

EXAMINATION ANSWER KEY

2017 SRO NRC Exam

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455828
User-Defined ID:	
Cross Reference Number:	
Topic:	T-102 Torus level blowdown
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.41(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-102, T-112	
Learning Objective:	PLOT-PBIG-2102 9b	
K/A System:	295030 Low Suppression Pool Wtr Lvl	Importance; SRO 4.2
K/A Statement:	A2.01 - Ability to determine and/or interpret the following as they apply to Low Suppression Pool Wtr Lvl: Suppression pool level	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

2 **ID: 1455488** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Annunciator 217 F-5, REACT BLDG COOLING WATER SUPPLY LO PRESS, alarms.
- RBCCW pressure is 65 psig and fluctuating.
- The "A" RBCCW pump is operating.
- The "B" RBCCW pump CANNOT be started.
- Annunciator 217 E-5, REACT BLDG COOLING WATER SUPPLY HI TEMP, alarms.
- RBCCW supply temperature is 120°F and rising slowly.
- There is NO evidence of system leakage.
- An operator in the field reports that the "A" RBCCW pump sounds like it is experiencing severe cavitation.

Which one of the following describes the required control of the plant in accordance with ON-113, Loss of RBCCW?

A. NO Reactor power reduction or securing of equipment is yet required.

B. Shutdown the running RWCU pump(s) and lower Reactor power per GP-9-2, Fast Reactor Power Reduction.

C. Lower Reactor power per GP-9-2, Fast Reactor Power Reduction, and trip one Recirc pump.

D. Scram the Reactor per GP-4, Manual Reactor Scram, and trip both Recirc pumps.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	ARC 217 F-5 requires entry into ON-113 based on low RBCCW pressure and failure of standby pump to restore pressure. Since the standby pump is unavailable, the running pump is operating abnormally, and system temperature is high and degrading, restoration of RBCCW is not imminent. Therefore, ON-113 step 2.2 requires shutting down the running RWCU pumps and lowering Reactor power per GP-9-2.
Distracters:	A	Since conditions show degradation of RBCCW and restoration is not imminent, ON-113 requires securing RWCU pumps and lowering power. Plausible because if conditions were given that system pressure restoration was imminent, then no securing of equipment or power reduction would be required.
	C	No indications are given for degraded Recirc pump temperatures, therefore tripping of a Recirc pump is not required. Plausible because tripping of RWCU pumps is required to reduce heat load and tripping one Recirc pump is possible and would also reduce heat load. Also plausible because if a Recirc pump experienced a high temperature, then tripping of the pump would become necessary per ON-113.
	D	Scramming the Reactor is not required. Plausible because ON-113 step 2.6 provides this exact direction if it becomes necessary to shutdown both Recirc pumps due to high temperatures.

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455488
User-Defined ID:	
Cross Reference Number:	295018 A2.05
Topic:	ON-113 - Degraded pressure
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC 217 F-5, ON-113	
Learning Objective:	PLOT-5035-9e	
K/A System:	295018 Partial or Total Loss of CCW	Importance; SRO 2.9
K/A Statement:	A2.05 - Ability to determine and/or interpret the following as they apply to Partial or Total Loss of CCW: System pressure	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

3 **ID: 1455547** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- SE-16, Grid Emergency, has been entered due to grid voltage disturbance.
- Newlinville 220-34 Line (343SU) is below the minimum required voltage for operability.
- An operator in the field has determined the following Diesel Generator fuel storage tank levels:

Diesel Generator	Fuel Storage Tank Level (gallons)
E1	34,500
E2	37,000
E3	28,000
E4	32,000

Which one of the following describes the ***most limiting*** Technical Specification requirement for restoring an offsite circuit and/or Diesel Generator to operable status?

A maximum of...

A. 14 days are allowed to restore a Diesel Generator to operable status.

B. 7 days are allowed to restore an offsite circuit to operable status.

C. 12 hours are allowed to restore an offsite circuit or a Diesel Generator to operable status.

D. 2 hours are allowed to restore a Diesel Generator to operable status.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	With Newlinville Line below the minimum voltage for operability, one of the two qualified offsite circuits are inoperable. Tech Spec 3.8.1 Required Action A.3 requires restoring the offsite circuit to operable status (7 days). *E4 fuel oil level being <33,000 gallons but >29,500 gallons results in Tech Spec 3.8.3 Required Action A.1 requiring restoration of fuel oil level to within limits (48 hours), but not declaring this DG inoperable yet. *E3 fuel oil level being <29,500 gallons results in Tech Spec 3.8.3 Required Action F.1 requiring declaration of *E3 inoperable immediately. Then, Tech Spec 3.8.1 Required Action B.5 requires restoring DG to operable status (14 days). The combination of Newlinville line inoperability and *E3 inoperability result in Tech Spec 3.8.1 Required Action E.1(2) requiring restoration of either the offsite circuit or DG to operable status (12 hours). This 12 hour requirement is the most limiting for restoration of operable status.
Distracters:	A	A more limiting 12 hour requirement exists because both a required offsite line is inoperable and one DG is inoperable. Plausible because Tech Spec 3.8.3 Required Action F.1 does currently require declaring *E3 inoperable immediately and Tech Spec 3.8.1 Required Action B.5 requires restoring *E3 to operable status (14 days).
	B	A more limiting 12 hour requirement exists because both a required offsite line is inoperable and one DG is inoperable. Plausible because Tech Spec 3.8.1 Required Action A.3 does currently require restoring the offsite circuit to operable status (7 days).
	D	No 2 hour requirement is currently applicable. Plausible because if a *E4 fuel oil level was lower (<29,500 gallons), such that 2 DGs were inoperable, then Tech Spec 3.8.1 Required Action F.1 would require restoring one of the DGs to operable status within a maximum of 2 hours.

Question 3 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455547
User-Defined ID:	
Cross Reference Number:	
Topic:	TS Call for Line + DG inoperable
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation

Source:	New Exam item **X** Modified Bank **(994164)** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	Tech Specs 3.8.1 and 3.8.3 and bases	
Learning Objective:	PLOT-5052-14	
K/A System:	700000 Generator Voltage and Electric Grid Disturbances	Importance; SRO 4.3
K/A Statement:	A2.09 - Ability to determine and/or interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of emergency diesel generators	
REQUIRED MATERIALS:	**Tech Specs 3.8.1 and 3.8.3**	
Notes and Comments:		

4 **ID: 1659149** **Points: 1.00**

Unit 2 is shutdown with the following:

- The "A" RHR pump is in Shutdown Cooling.
- Reactor coolant temperature is 135°F.
- The Reactor head is still installed.
- The "B" Recirculation pump is operating and the "A" Recirculation pump is shutdown.

Then, MO-2-10-25A, Inboard Discharge, fails closed.

Which of the following describes the number of operable RHR Shutdown Cooling subsystems and the required control of the "A" Recirculation pump?

The number of operable RHR Shutdown Cooling subsystems is __(1)__.

Starting the "A" Recirculation pump __(2)__ required.

A. (1) only one
(2) is

B. (1) only one
(2) is NOT

C. (1) two
(2) is

D. (1) two
(2) is NOT

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	TS 3.4.8 applies because the plant is in Mode 4. TS 3.4.8 requires two operable SDC subsystems. TS 3.4.8 bases state that each RHR loop is considered to have two subsystems. The given failure makes all of RHR loop A inoperable for SDC, but leaves all of RHR loop B operable. Therefore, two subsystems are operable with just RHR loop B. The A Recirculation pump is not required to be started. Plausible because with delays in restoring SDC, if the B Recirculation pump were not running, a Recirculation pump would need to be started. Also plausible because SDC B will be placed in service and it is typical to operate the opposite Recirculation pump.
Distracters:	A	TS 3.4.8 bases state that each RHR loop is considered to have two subsystems, so RHR loop B is still providing the required operable subsystems. Plausible that RHR loop B would only be considered one subsystem due to common components. The A Recirculation pump is not required to be started. Plausible because with delays in restoring SDC, if the B Recirculation pump were not running, a Recirculation pump would need to be started. Also plausible because SDC B will be placed in service and it is typical to operate the opposite Recirculation pump.
	B	TS 3.4.8 bases state that each RHR loop is considered to have two subsystems, so RHR loop B is still providing the required operable subsystems. Plausible that RHR loop B would only be considered one subsystem due to common components.
	C	The A Recirculation pump is not required to be started. Plausible because with delays in restoring SDC, if the B Recirculation pump were not running, a Recirculation pump would need to be started. Also plausible because SDC B will be placed in service and it is typical to operate the opposite Recirculation pump.

Question 4 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659149
User-Defined ID:	
Cross Reference Number:	
Topic:	TS 3.4.8 required subsystems
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	TS 3.4.8 and bases	
Learning Objective:	PLOT-5010-15	
K/A System:	295021 Loss of Shutdown Cooling	Importance; SRO 4.2
K/A Statement:	2.2.25 - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

5 **ID: 1455906** **Points: 1.00**

Unit 2 is operating at 30% power with the following:

- Vibration levels on the high pressure Turbine are rising.
- The following vibration levels are indicated:
  - VBE-20S772-01XD: 6 mils
  - VBE-20S772-01YD: 4 mils
  - VBE-20S772-02XD: 6 mils
  - VBE-20S772-02YD: 4 mils
  - All of these indications are rising at a rate of 0.5 mils/minute.

Note: Assume the vibration rate of rise remains constant and no further power change occurs.

Given the following portion of a Turbine vibration alarm table:

VIBRATION CHANNELS				
PROBE	HI ALARM SETPOINT	HI-HI ALARM SETPOINT	VR-2657 RECORDER POINT	DAS DISPLAY
VBE-20S772-01XD (HP)	7 mils	10 mils	1	XI-80838
VBE-20S772-01YD (HP)	5 mils	8 mils	none	XI-80838
VBE-20S772-02XD (HP)	7 mils	10 mils	2	XI-80838
VBE-20S772-02YD (HP)	5 mils	8 mils	none	XI-80838
VBE-2AS773-03XD (LP A)	7 mils	11 mils	3	XI-80838
VBE-2AS773-03YD (LP A)	7 mils	10 mils	none	XI-80838
VBE-2AS773-04XD (LP A)	7 mils	11 mils	4	XI-80838
VBE-2AS773-04YD (LP A)	7 mils	10 mils	none	XI-80838
VBE-2BS773-05XD (LP B)	7 mils	11 mils	5	XI-80838
VBE-2BS773-05YD (LP B)	7 mils	10 mils	none	XI-80838
VBE-2BS773-06XD (LP B)	7 mils	11 mils	6	XI-80838
VBE-2BS773-06YD (LP B)	7 mils	10 mils	none	XI-80838
VBE-2CS773-07XD (LP C)	7 mils	11 mils	7	XI-80838
VBE-2CS773-07YD (LP C)	7 mils	10 mils	none	XI-80838
VBE-2CS773-08XD (LP C)	7 mils	11 mils	8	XI-80838
VBE-2CS773-08YD (LP C)	7 mils	10 mils	none	XI-80838

Which one of the following describes when the Turbine will be required to be tripped, in accordance with the Alarm Response Cards, and whether a Reactor scram is also required?

The Turbine will first be required to be tripped in...

A. 2 minutes. A Reactor scram will also be required.

B. 2 minutes. A Reactor scram will NOT also be required.

C. 8 minutes. A Reactor scram will also be required.

D. 8 minutes. A Reactor scram will NOT also be required.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	ARC 205 A-2 and B-2 provide direction for response to high Turbine vibrations. Annunciator 205 A-2 will first be received in 2 minutes when the Hi alarm setpoints are exceeded for the given vibration points. This ARC discusses the Turbine trip, but does not require it until the Hi-Hi alarm setpoints are reached (which is when Annunciator 205 B-2 alarms). The Hi-Hi alarm setpoints will be reached in 8 minutes. With Reactor power at 30%, and assuming no further power change, a Reactor scram is also required because the Turbine Stop Valve closure scram is active with Reactor power greater than 26.7%. **Note:** The question meets SRO level guidelines because the timing of the Turbine trip requires assessment of conditions (vibrations levels and rate of rise) and selecting when to implement a section of a procedure (Turbine trip requirement in ARCs for high Turbine vibration). The question cannot be answered solely be knowing systems knowledge, immediate operator actions, AOP/EOP entry conditions, or overall mitigative strategy.
Distracters:	A	The Turbine trip is not required for 8 minutes. Plausible because in 2 minutes, the Hi alarm setpoint will be reached, which brings in Annunciator 205 A-2, and ARC 205 A-2 includes the Turbine trip requirement. Also plausible the Turbine trip would be at the Hi alarm setpoint and breaking vacuum would be at the Hi-Hi alarm setpoint.
	B	The Turbine trip is not required for 8 minutes. Plausible because in 2 minutes, the Hi alarm setpoint will be reached, which brings in Annunciator 205 A-2, and ARC 205 A-2 includes the Turbine trip requirement. Also plausible the Turbine trip would be at the Hi alarm setpoint and breaking vacuum would be at the Hi-Hi alarm setpoint. A Reactor scram will be required because the Turbine Stop Valve closure scram is active with Reactor power greater than 26.7%. Plausible because Reactor power is low, and if it were lower, then only a Turbine trip would be required.
	D	A Reactor scram will be required because the Turbine Stop Valve closure scram is active with Reactor power greater than 26.7%. Plausible because Reactor power is low, and if it were lower, then only a Turbine trip would be required.

Question 5 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455906
User-Defined ID:	
Cross Reference Number:	
Topic:	Turbine vibrations - Trip/Scram requirements
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ARC 205 A-2 and B-2, ARC 210 A-1	
Learning Objective:	PLOT-5001B 8a	
K/A System:	295006 SCRAM	Importance; SRO 4.2
K/A Statement:	2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

6 **ID: 1455604** **Points: 1.00**

Unit 2 is operating at 100% power when a Loss of Instrument Air results in the following:

- Annunciator 211 D-2, SCRAM VALVE PILOT AIR HEADER PRESS HI-LOW, alarms.
- Annunciator 216 D-3, A INSTRUMENT AIR HEADER LO PRESS, alarms.
- Annunciator 216 D-4, B INSTRUMENT AIR HEADER LO PRESS, alarms.
- Scram air header pressure is 50 psig and lowering.
- Annunciator 211 D-4, ROD DRIFT, alarms.
- The URO reports control rod 22-23 is drifting in.

Which one of the following actions is required for these conditions?

A. Scram the Reactor now per ON-119, Loss of Instrument Air.

B. Scram the Reactor only if a second control rod drifts per ON-121, Drifting Control Rod.

C. Use the EMER IN control switch to insert rod 22-23 to Full-In per ON-121, Drifting Control Rod.

D. Begin a rapid plant shutdown using GP-9-2, Fast Reactor Power Reduction, per ON-119, Loss of Instrument Air.

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	ON-119 entry is required based on IA System alarms. ON-119 directs a reactor scram if any control rod begins to drift in due to decreasing scram air header pressure. The given conditions indicate that scram air header pressure is lowering.
Distracters:	B	This is the correct action per ON-121 for a second drifting control rod, but is overridden by the direction in ON-119 to scram on the first drifting rod.
	C	This is the correct action per ON-121 for a drifting control rod only (i.e., NOT due to a loss of instrument air). Entry into ON-119 (and direction to scram) overrides ON-121 actions for a drifting control rod.
	D	This is required by ON-119 when instrument air header pressure cannot be stabilized above 75 psig, but is overridden by the requirement to scram if any control rod begins to drift.

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455604
User-Defined ID:	
Cross Reference Number:	2.1.7
Topic:	ON-119 Scram on drifting rod
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

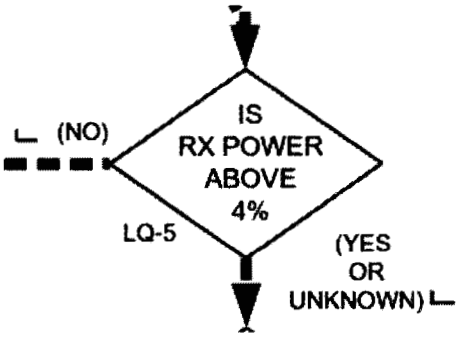
Source Documentation

Source:	New Exam item Exam **(2013 NRC #80)** Modified Bank Exam Bank **X** ILT Exam Bank	**X** Previous NRC Other
Reference(s):	ON-119, ON-121	
Learning Objective:	PLOT-DBIG-1550-22a	
K/A System:	295019 Partial or Total Loss of Inst. Air	Importance; SRO 4.7
K/A Statement:	2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

7 **ID: 1455678** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- The Main Turbine trips.
- Control rods are NOT all full in.
- All control rods are inserted to at least position 12.
- All WRNM and APRM indications are lost.
- Reactor pressure is 1100 psig, steady.
- Two (2) SRVs are open.
- MSIVs are closed.
- HPCI and RCIC are in standby.
- T-117, Level/Power Control, is being executed.
- No actions have yet been taken to reduce Reactor power.
- The next step in T-117 is:



Which one of the following describes how this step is required to be answered in accordance with T-117 and the associated bases?

Answer...

A. "NO" based on the control rod pattern.

B. "NO" based on SRV operation and Reactor pressure response.

C. "YES OR UNKNOWN" based on the control rod pattern.

D. "YES OR UNKNOWN" based on SRV operation and Reactor pressure response.

Answer: D

Answer Explanation		
Choice		Basis or Justification
Correct:	D	T-117 bases state that this Reactor power determination does not require use of APRMs, but can be made using other alternate indications as well. Each SRV passes steam flow equivalent to approximately 4.9% Reactor power. With 2 SRVs open and Reactor pressure stable at 1100 psig, Reactor power is above 4%. **Note:** The question meets the K/A by presenting a situation with high Reactor pressure (1100 psig and 2 SRVs open) and requiring the candidate to interpret the plant response to these conditions relative to current Reactor power vs. normal decay heat generation.
Distracters:	A	Control rod pattern is not used in this determination. Plausible because control rod pattern is used in other T-101/T-117 steps regarding whether an ATWS is in progress or not, and all control rods being inserted past a certain bank position can be used to answer this question "NO". Position 12 is plausible as it is analogous to the MAXIMUM SUBCRITICAL BANKED WITHDRAWAL POSITION of 02 used in this determination and all rods in past 12 is a significant amount of negative reactivity.
	B	Each SRV passes steam flow equivalent to approximately 4.9% Reactor power. With 2 SRVs open and Reactor pressure stable at 1100 psig, Reactor power is above 4%. Plausible if candidate though 2 SRVs open at 1100 psig following a scram was a normal indication for decay heat generation and allowed answering the question "NO".
	C	Control rod pattern is not used in this determination. Plausible because control rod pattern is used in other T-101/T-117 steps regarding whether an ATWS is in progress or not, and all control rods being inserted past a certain bank position can be used to answer this question.

Question 7 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455678
User-Defined ID:	
Cross Reference Number:	295025 A2.05
Topic:	Power determination without APRMs
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank **(SSES LOC26R NRC #89)** Other Exam Bank ILT Exam Bank	
Reference(s):	T-117 and bases, PLOT5001A	
Learning Objective:	PLOT-PBIG-2117 7b	
K/A System:	295025 High Reactor Pressure	Importance; SRO 3.6
K/A Statement:	A2.05 - Ability to determine and/or interpret the following as they apply to High Reactor Pressure: Decay heat generation	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

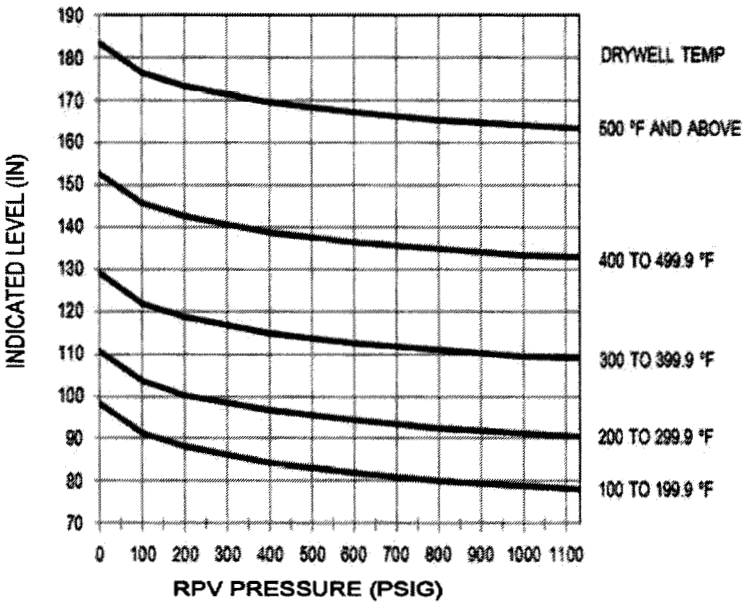
8 **ID: 1659201** **Points: 1.00**

Unit 2 has experienced a scram with the following:

- Turbine Bypass Valves have all failed closed.
- Reactor pressure is 1050 psig and slowly rising.
- Narrow Range level indicators (LI-2-06-094) are upscale.
- Wide Range level indicators (LI-2-02-6-085) indicate +60" and steady.
- Shutdown Range indicator (LI-2-2-3-86) indicates +85" and steady.
- Drywell temperature is 140°F and steady.
- Instrument nitrogen has been restored to the Drywell.

Note: OT-110, Reactor High Level, Figure 1 is provided below.

FIGURE 1



Which one of the following identifies the required control of MSIVs and SRVs, in accordance with OT-110?

MSIVs...

A. must be closed. Reduce Reactor pressure to less than 850 psig using prolonged operation of a single SRV.

B. must be closed. Maintain SRVs closed unless Reactor pressure reaches the pressure relief setpoint.

C. may remain open. Maintain SRVs closed unless Reactor pressure reaches the pressure relief setpoint.

D. may remain open. Reduce Reactor pressure to less than 850 psig using prolonged operation of a single SRV.

Answer: A

Answer Explanation		
Choice	Basis or Justification	
Correct	A	With high Reactor water level, Shutdown Range indicator (LI-2-2-3-86) must be used. This instrument is calibrated for shutdown conditions, therefore OT-110 Figure 1 must be used to interpret Reactor water level with respect to the Main Steam Lines. For the given conditions, level is above the bottom of the Main Steam Lines (which are at an actual level of +108", but an indicated level of about +78"). With Reactor water level above the bottom of the Main Steam Lines, OT-110 requires closing the MSIVs. Additionally, with Reactor pressure at 1050 psig, OT-110 requires reducing Reactor pressure to less than 850 psig using prolonged operation of a single SRV.
Distractors	B	OT-110 requires reducing Reactor pressure to less than 850 psig using prolonged operation of a single SRV. Plausible because OT-110 requires keeping SRVs closed unless Reactor pressure reaches 1050 psig while Main Steam Lines are still flooded and indicated Reactor water level is below the actual 108" height of the bottom of the Main Steam Lines.
	C	OT-110 requires closing the MSIVs. Plausible because indicated Reactor water level is below the actual 108" height of the bottom of the Main Steam Lines.
	D	OT-110 requires closing the MSIVs. Plausible because indicated Reactor water level is below the actual 108" height of the bottom of the Main Steam Lines. OT-110 requires reducing Reactor pressure to less than 850 psig using prolonged operation of a single SRV. Plausible because OT-110 requires keeping SRVs closed unless Reactor pressure reaches 1050 psig while Main Steam Lines are still flooded and indicated Reactor water level is below the actual 108" height of the bottom of the Main Steam Lines.

Question 8 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1659201
User-Defined ID:	
Cross Reference Number:	
Topic:	OT-110 MSIV and SRV control
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

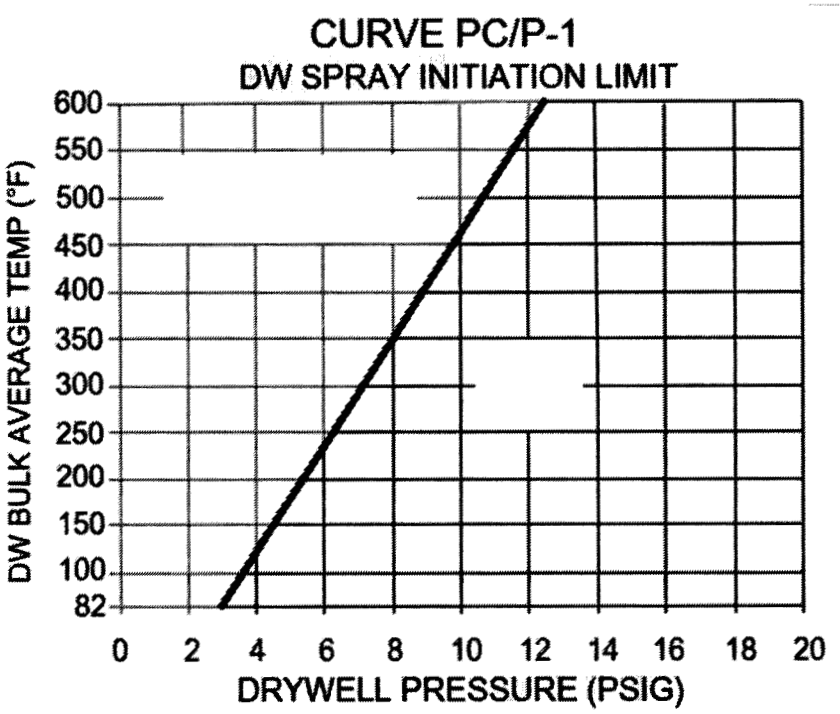
Source:	New Exam item **X** Modified Bank **(2009 Cert #85)** ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	OT-110	
Learning Objective:	PLOT-DBIG-1540 7	
K/A System:	295008 High Reactor Water Level	Importance; SRO 3.9
K/A Statement:	A2.01 - Ability to determine and/or interpret the following as they apply to High Reactor Water Level: Reactor water level	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

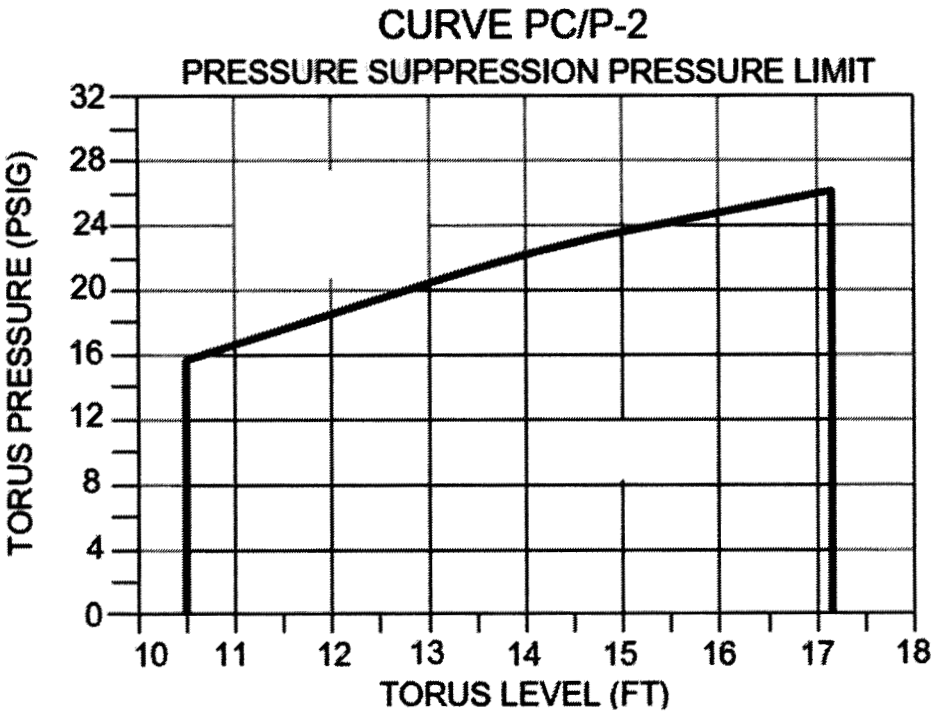
9 **ID: 1455907** **Points: 1.00**

A loss of coolant accident has resulted in the following on Unit 2:

- Reactor water level is -30" and slowly rising.
- Reactor pressure is 700 psig and slowly lowering.
- Drywell pressure is 25 psig and slowly rising.
- Drywell temperature is 250°F and slowly rising.
- Torus pressure is 22 psig and slowly rising.
- Torus water level is 15' and stable.
- Torus spray has just been initiated.
- No other actions have yet been directed from T-102, Primary Containment Control.

Note: The Drywell Spray Initiation Limit and Pressure Suppression Pressure Limit curves are provided below.





Which one of the following describes the required action(s), if any, in accordance with T-102?

A. Do NOT initiate Drywell spray. Do NOT perform an Emergency Blowdown.

B. Do NOT initiate Drywell spray. Perform an Emergency Blowdown per T-112.

C. Initiate Drywell spray. Also concurrently perform an Emergency Blowdown per T-112.

D. Initiate Drywell spray. Do NOT concurrently perform an Emergency Blowdown per T-112.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	Drywell spray is required because Torus level is below 18', the DW Spray Initiation Limit curve is satisfied, and Torus pressure is above 9 psig. The Pressure Suppression Pressure Limit curve is currently being approached, but is not yet violated. The mitigating strategy of T-102 allows attempting sprays and then assessing the effectiveness of sprays before deciding if the Pressure Suppresion Pressure Limit can be maintained or not. Therefore, an Emergency Blowdown is not required to be performed concurrently with initiating sprays. The need for an Emergency Blowdown is assessed after sprays are in service.
Distracters:	A	Drywell spray should be initiated. Plausible because Torus level is high (above 18' would preclude initiating Drywell spray) and Drywell temperature is high (at lower pressures, DWSIL curve could be challenged).
	B	Drywell spray should be initiated. Plausible because Torus level is high (above 18' would preclude initiating Drywell spray) and Drywell temperature is high (at lower pressures, DWSIL curve could be challenged).
	C	The Pressure Suppression Pressure Limit curve is currently being approached, but is not yet violated. The mitigating strategy of T-102 allows attempting sprays and then assessing the effectiveness of sprays before deciding if the Pressure Suppression Pressure Limit can be maintained or not. Therefore, an Emergency Blowdown is not required to be performed concurrently with initiating sprays. The need for an Emergency Blowdown is assessed after sprays are in service. Plausible because the Pressure Suppression Pressure Limit curve is currently being approached and will be violated if sprays are ineffective, which would eventually require Emergency Blowdown. Additionally, a Torus pressure of 22 psig would violate the Pressure Suppression Pressure Limit at lower Torus water levels.

Question 9 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455907
User-Defined ID:	
Cross Reference Number:	2.4.6
Topic:	When to spray and when to assess PSP
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item NRC Exam **(JAF 9/14 NRC #78)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	T-102 and bases	
Learning Objective:	PLOT-PBIG-2102 9b	
K/A System:	295010 High Drywell Pressure	Importance; SRO 4.7
K/A Statement:	2.4.6 - Knowledge of EOP mitigation strategies.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

10 **ID: 1455691** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- The A CRD pump has tripped due to a motor fault.
- A breaker issue is preventing start of the B CRD pump.
- Maintenance is troubleshooting the B CRD pump breaker issue.
- Annunciator 211 G-5, CRD HYDRAULIC HI TEMP, alarms.
- Control rod 18-31 temperature is 300°F and rising slowly.
- Control rod 18-31 is at position 48.

Which one of the following describes the required action for the control rod high temperature in accordance with ON-107, Loss of CRD Regulating Function?

Scram...

A. the Reactor now.

B. the Reactor if flow is NOT returned to normal within one hour.

C. control rod 18-31 using the individual rod scram test switch now.

D. control rod 18-31 using the individual rod scram test switch if flow is NOT returned to normal within one hour.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	ON-107 step 2.9 states, "IF any CRD High Temperature alarm is verified to be valid on withdrawn Control Rods AND CRD Hydraulic System flow is NOT returned to normal within one hour of the first alarm, THEN SCRAM AND ENTER T-100, "Scram" OR T-101, "RPV Control" as appropriate.
Distracters:	A	The Reactor is only scrammed if flow is not returned to normal within one hour. Plausible because ON-107 has other immediate scram requirements and the high temperature does risk permanent damage to the CRD seals that could affect the scram function.
	C	No immediate scram is required, and if a scram does become required, it is for all control rods, not just the rod with the high temperature. Plausible because ON-107 has other immediate scram requirements and the temperature issue is only with the one control rod. If a CRD pump were running, the single rod would be given a continuous withdrawal signal to help in cooling.
	D	If a scram does become required, it is for all control rods, not just the rod with the high temperature. Plausible because the temperature issue is only with the one control rod. If a CRD pump were running, the single rod would be given a continuous withdrawal signal to help in cooling.

Question 10 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455691
User-Defined ID:	
Cross Reference Number:	295022 A2.03
Topic:	ON-107 high CRD temperature
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ON-107	
Learning Objective:	PLOT-PBIG-1550 12a	
K/A System:	295022 Loss of CRD Pumps	Importance; SRO 3.2
K/A Statement:	A2.03 Ability to determine and/or interpret the following as they apply to Loss of CRD Pumps: CRD mechanism temperatures	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

11 **ID: 1455909** **Points: 1.00**

Unit 2 has experienced a loss of coolant accident with the following:

- Reactor water level is -160" and lowering slowly.
- Reactor pressure is 650 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- All RHR pumps have started.
- All Core Spray pumps have failed to start.
- ADS has NOT been inhibited.
- The following annunciators came into alarm 10 seconds ago:
  - 227 D-4, Blowdown Timers Initiated
  - 227 E-4, Blowdown Aux Relays Energized RHR or CS

Which one of the following describes:

(1) the response of the ADS valves if the current conditions continue until the ADS timers time out, and

(2) the required control of ADS given the initial conditions, in accordance with the Emergency Operating Procedures?

A. (1) ADS valves will open.
(2) Inhibit ADS.

B. (1) ADS valves will open.
(2) Allow ADS valves to open.

C. (1) ADS valves will remain closed.
(2) Inhibit ADS.

D. (1) ADS valves will remain closed.
(2) Manually open ADS valves.

Answer: A

Answer Explanation

Choice		Basis or Justification
Correct:	A	A valid ADS signal exists due to Reactor water level lowering to -160". When the timer times out, the ADS valves will open because sufficient RHR pumps are running to satisfy the logic. T-111 requires inhibiting ADS in this situation. Emergency Blowdown is intentionally delayed until Reactor water level reaches -172".
Distracters:	B	T-111 requires inhibiting ADS in this situation. Emergency Blowdown is intentionally delayed until Reactor water level reaches -172". Plausible because allowing ADS valves to open would lower Reactor pressure and allow RHR to restore Reactor water level. This will be the strategy once Reactor water level reaches -172".
	C	ADS valves will open. Plausible because all Core Spray pump have failed.
	D	ADS valves will open. Plausible because all Core Spray pump have failed. ADS valves are not manually opened unless Reactor water level reaches -172". Plausible to open the valves if the automatic action failed.

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455909
User-Defined ID:	
Cross Reference Number:	218000 A2.06
Topic:	Control of ADS with low level
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation		
Source:	**X** New Exam item NRC Exam **(JAF 2016 NRC #87)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	T-101, T-111, ARC-227 D-4 and E-4	
Learning Objective:	PLOT-5001G 10f	
K/A System:	218000 ADS	Importance; SRO 4.3
K/A Statement:	A2.06 - Ability to (a) predict the impacts of the following on the ADS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS initiation signals present	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

12 **ID: 1476313** **Points: 1.00**

Unit 2 is operating at 100% power when Electrical Maintenance reports that 480 VAC Load Center E-324 must be de-energized due to a major hot spot on the bus.

Which one of the following describes the maximum time for restoring the bus to operable status before needing to enter a shutdown action in accordance with Technical Specifications?

A. 2 hours

B. 8 hours

C. 7 days

D. 14 days

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	480 VAC Load Center E-324 is required per Tech Spec Bases Table B 3.8.7-1 for Unit 2. With the bus de-energized, Tech Spec 3.8.7 Condition C must be entered. The Required action is to restore the bus to operable status within a maximum of 8 hours, or then enter Condition E, which requires being in Mode 3 within 12 hours
Distracters :	A	The maximum time is 8 hours, not 2 hours. Plausible because 2 hours is the correct answer for a required DC bus per TS 3.8.7 Condition D.
	C	The maximum time is 8 hours, not 7 days. Plausible because multiple related TS have 7 day completion times for restoration of equipment, such as TS 3.8.1 Condition A for restoring an inoperable offsite power source.
	D	The maximum time is 8 hours, not 14 days. Plausible because multiple related TS have 14 day completion times for restoration of equipment, such as TS 3.8.1 Condition B for restoring an inoperable DG.

Question 12 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1476313
User-Defined ID:	
Cross Reference Number:	262001 A2.06
Topic:	TS 3.8.7 for deenergizing E324
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation		
Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	TS 3.8.7 and bases	
Learning Objective:	PLOT-5054 10f	
K/A System:	262001 AC Electrical Distribution	Importance; SRO 2.9
K/A Statement:	A2.06 - Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Deenergizing a plant bus	
REQUIRED MATERIALS:	**TS 3.8.1 and TS 3.8.7**	
Notes and Comments:		

13 **ID: 993101** **Points: 1.00**

Unit 2 is operating at 100% power when one of the two required "A" Main Steam Line flow channels is declared inoperable.

Which one of the following describes the Technical Specification requirement(s)?

A. An RPS Trip System must be tripped within a maximum of 12 hours, only.

B. A PCIS Group I Channel must be tripped within a maximum of 24 hours, only.

C. An RPS Trip System and a PCIS Group I Channel must both be tripped within a maximum of 12 hours.

D. An RPS Trip system must be tripped within a maximum of 12 hours and a PCIS Group I Channel must be tripped within a maximum of 24 hours.

Answer: B

Answer Explanation

Choice		Basis or Justification
Correct:	B	There are two required MSL flow channels for each MSL. With one of the two required channels declared inoperable, Tech Spec Table 3.3.6.1-1 Function 1.c is not fully met, therefore Condition A applies. Tech Spec 3.3.6.1 Condition A requires placing the PCIS channel in trip within a maximum of 24 hours for this specific Function.
Distracters:	A	This instrument does not directly require TS 3.3.1.1 entry and placing an RPS Trip System in trip. Plausible because TS 3.3.1.1 and 3.3.6.1 share some instruments, such as Reactor water level, Drywell pressure, and up until a recent change, MSL radiation. Also plausible because this failure affects MSL isolation capability, which in turn could affect whether the MSIV closure scram functions when required.
	C	This instrument does not directly require TS 3.3.1.1 entry and placing an RPS Trip System in trip. Plausible because TS 3.3.1.1 and 3.3.6.1 share some instruments, such as Reactor water level, Drywell pressure, and up until a recent change, MSL radiation. Also plausible because this failure affects MSL isolation capability, which in turn could affect whether the MSIV closure scram functions when required. 12 hours for PCIS here is plausible because for shared instruments, TS 3.3.6.1 Condition A has a more limiting 12 hour time requirement.
	D	This instrument does not directly require TS 3.3.1.1 entry and placing an RPS Trip System in trip. Plausible because TS 3.3.1.1 and 3.3.6.1 share some instruments, such as Reactor water level, Drywell pressure, and up until a recent change, MSL radiation. Also plausible because this failure affects MSL isolation capability, which in turn could affect whether the MSIV closure scram functions when required.

Question 13 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	1.00
System ID:	993101
User-Defined ID:	SRO-ILT-5007G-7-001
Cross Reference Number:	223002 2.2.22
Topic:	SRO-ILT-5007G-7-001 T/S inoperable MSL flow channel
Num Field 1:	2931
Num Field 2:	A NRC
Text Field:	A

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation

Source:	New Exam item Exam **X** Modified Bank **(368323)** Exam Bank ILT Exam Bank	Previous NRC Other
Reference(s):	TS 3.3.1.1 and 3.3.6.1	
Learning Objective:	PLOT-5007G 13	
K/A System:	223002 PCIS/Nuclear Steam Supply Shutoff	Importance; SRO 4.7
K/A Statement:	2.2.22 - Knowledge of limiting conditions for operations and safety limits.	
REQUIRED MATERIALS:	**TS 3.3.1.1 and 3.3.6.1 (with allowable values removed)**	
Notes and Comments:		

14 **ID: 1455728** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Annunciator 211 H-3, STANDBY LIQUID SQUIB VALVE LOSS OF CONTINUITY, alarms.
- The A squib valve continuity light is lit.
- The B squib valve continuity light is extinguished.
- The A squib valve ammeter indicates 5 mA.
- The B squib valve ammeter indicates 0 mA.

Which one of the following describes the most restrictive action required for this failure in Technical Specification (TS) 3.1.7, Standby Liquid Control (SLC) System, if any?

A. TS 3.1.7 Condition entry is NOT required.

B. Be in Mode 3 within a maximum of 12 hours.

C. Restore one SLC subsystem to operable status within a maximum of 8 hours.

D. Restore one SLC subsystem to operable status within a maximum of 7 days.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The given indications show that the B squib valve is inoperable. This requires declaring one SLC subsystem inoperable and entering TS 3.1.7 Condition B, only. This Condition requires restoring one SLC subsystem to operable status within a maximum of 7 days.
Distracters:	A	The B squib valve will still operate properly, which means both the A and B SLC pumps can still inject to the Reactor. However, TS 3.1.7 bases still require declaring one subsystem inoperable and entering TS 3.1.7 Condition B for this failure. Plausible because both SLC pumps can still injection boron to the Reactor with the one operable squib valve.
	B	TS 3.1.7 Condition B requires restoring one system to operable within a maximum of 7 days. Plausible because TS 3.1.7 Condition D requires being in Mode 3 within a maximum of 12 hours under different circumstances.
	C	TS 3.1.7 Condition B requires restoring one system to operable within a maximum of 7 days, not 8 hours. Plausible because TS 3.1.7 Condition C requires restoring one system to operable within a maximum of 8 hours under different circumstances and candidate could believe both subsystems are inoperable because both SLC pumps lose one of their two normal injection valves.

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455728
User-Defined ID:	
Cross Reference Number:	211000 2.4.45
Topic:	TS 3.1.7 - One inop squib valve
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation

Source:	New Exam item **X** Modified Bank **(992994)** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	ARC-211-H-3, TS 3.1.7 and bases	
Learning Objective:	PLOT-5011 13	
K/A System:	211000 SLC	Importance; SRO 4.3
K/A Statement:	2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.	
REQUIRED MATERIALS:	**TS 3.1.7**	
Notes and Comments:		

15 **ID: 1683202** **Points: 1.00**

Unit 2 has experienced a failure to scram with the following:

- T-117, Level/Power Control, is being executed.
- Reactor power is downscale on all APRMs.
- SBLC has NOT been injected into the Reactor.
- Reactor pressure is 300 psig and stable on Turbine Bypass Valves.
- Reactor water level is +20" and slowly lowering.
- Condensate pump A is injecting through AO-8091 using the startup level control station in manual.

Then, the air supply to AO-8091 is lost. NO other valves are affected.

Which one of the following (1) describes the effect on Condensate flow to the Reactor and (2) identifies the ability to use Core Spray to augment injection to the Reactor under current conditions, in accordance with T-117?

A. (1) Flow lowers
(2) Core Spray injection is allowed

B. (1) Flow lowers
(2) Core Spray injection is NOT allowed

C. (1) Flow rises
(2) Core Spray injection is allowed

D. (1) Flow rises
(2) Core Spray injection is NOT allowed

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	Loss of air to AO-8091 causes the valve to fail open, resulting in rising flow to the Reactor. In the given conditions, T-117 does not allow injection with Core Spray. Core Spray injection is only allowed following Emergency Blowdown or after T-117 is exited.
Distracters:	A	Flow rises, not lowers. Plausible because this would be the response if AO-8091 failed closed, as do many other valves.
	B	Flow rises, not lowers. Plausible because this would be the response if AO-8091 failed closed, as do many other valves. Core Spray injection is not allowed because T-117 is being executed and the Reactor has not undergone Emergency Blowdown (as evidenced by pressure at 300 psig and stable on TBVs). Plausible because power is downscale, level is being controlled in the normal band, and Reactor pressure is low.
	C	Core Spray injection is not allowed because T-117 is being executed and the Reactor has not undergone Emergency Blowdown (as evidenced by pressure at 300 psig and stable on TBVs). Plausible because power is downscale, level is being controlled in the normal band, and Reactor pressure is low.

Question 15 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1683202
User-Defined ID:	
Cross Reference Number:	
Topic:	Loss of air to AO-8091, use of CS in T-117
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation		
Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ON-119, T-117	
Learning Objective:	PLOT-5006 10e	
K/A System:	259002 Reactor Water Level Control	Importance; SRO 3.4
K/A Statement:	A2.05 - Ability to (a) predict the impacts of the following on the Reactor Water Level Control; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of applicable plant air systems	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

16 **ID: 1588328** **Points: 1.00**

Both Units are operating at 100% power with the following:

- An uncontrolled release to the environment occurs.
- The Main Control Room air intake radiation monitors are indicating 2,400 cpm and stable.

Which one of the following describes (1) the resulting status of the Control Room Ventilation system and (2) the required control of the system or plant, in accordance with plant procedures?

A. (1) Remains in normal lineup
(2) Maintain the system in normal lineup

B. (1) Remains in normal lineup
(2) Manually place the system in Emergency mode

C. (1) Automatically shifts to Emergency mode
(2) If Control Room temperature reaches 90°F, then perform a rapid shutdown

D. (1) Automatically shifts to Emergency mode
(2) If respiratory equipment becomes required for Control Room staff for more than 2 hours, then perform a rapid shutdown

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The given Main Control Room air intake radiation levels are higher than normal and above the automatic initiation setpoint for the Control Room Ventilation Emergency mode of 350 cpm. Therefore, the Control Room Ventilation system does automatically shift to the Emergency Mode. This requires entry into ON-115, Loss of Normal Main Control Room Ventilation. ON-115 requires rapid shutdown of the plant if the need for use of respirators by Control Room staff exceeds 2 hours.
Distracters:	A	The Control Room Ventilation system automatically shifts to the Emergency Mode. Plausible if candidate believes a higher rad level is required or if system is only manually started in the event of high radiation.
	B	The Control Room Ventilation system automatically shifts to the Emergency Mode. Plausible if candidate believes a higher rad level is required or if system is only manually started in the event of high radiation.
	C	With high radiation conditions present, ON-115 requires rapid shutdown only if Control Room temperature reaches 100°F, not 90°F. Plausible because a rapid shutdown would become required if temperature rose further.

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588328
User-Defined ID:	
Cross Reference Number:	
Topic:	High rad start, ON-115 requirement
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics			
Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation		
Source:	New Exam item **X** Modified Bank **(993184)** Bank ILT Exam Bank	Previous NRC Exam Other Exam
Reference(s):	ARC-003 A-2, ON-115	
Learning Objective:	PLOT-5050D 10b	
K/A System:	290003 Control Room HVAC	Importance; SRO 3.4
K/A Statement:	A2.02 - Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Extreme environmental conditions	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

17 **ID: 1455729** **Points: 1.00**

Which one of the following identifies a radiation monitor used to define Loss of the Fuel Clad fission product barrier, in accordance with EP-AA-1007, Emergency Action Levels for Peach Bottom Atomic Power Station?

A. Drywell radiation monitor

B. Main Stack radiation monitor

C. Main Steam Line radiation monitor

D. Air Ejector discharge radiation monitor

Answer: A

Answer Explanation		
Choice		Basis or Justification
Correct:	A	The Drywell radiation monitor is used to define a Loss of Fuel Clad barrier in the EALs.
Distracters:	B	The Main Stack radiation monitor is not used to define a Loss of Fuel Clad barrier in the EALs. Plausible because this radiation monitor does provide evidence of fuel degradation and is used in other EALs (RG1, RS1, RA1, RU1).
	C	The Main Steam Line radiation monitor is not used to define a Loss of Fuel Clad barrier in the EALs. Plausible because this radiation monitor does provide evidence of fuel degradation and is used in ON-103 to determine when to scram the Reactor based on fuel degradation.
	D	The Air Ejector discharge radiation monitor is not used to define a Loss of Fuel Clad barrier in the EALs. Plausible because this radiation monitor does provide evidence of fuel degradation and is used in EAL RU3.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455729
User-Defined ID:	
Cross Reference Number:	272000 2.1.27
Topic:	Rad monitor for Loss of Fuel Clad barrier
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
MEMORY			10CFR55.43(b)(4)

Source Documentation

Source:	**X** New Exam item Previous NRC Exam Modified Bank **(JAF 9/14 NRC #100)** Other Exam Bank ILT Exam Bank	
Reference(s):	EP-AA-1007	
Learning Objective:	G6 5	
K/A System:	272000 Radiation Monitoring	Importance; SRO 4.0
K/A Statement:	2.1.27 - Knowledge of system purpose and/or function.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

18 **ID: 1588332** **Points: 1.00**

Both plants are operating at 100% power with the following:

- A seismic event occurs.
- Annunciator 316 A-5, OPERATING BASIS EARTHQUAKE EXCEEDED, alarms.
- A leak develops from the Unit 2 Reactor Water Cleanup (RWCU) system into the Reactor Building.
- Both automatic and manual isolation of Unit 2 RWCU have failed to stop the leak.
- Unit 2 Area temperature has exceeded the T-103, Secondary Containment Control, Action Level for the General Area 165' elevation.
- The Unit 2 Main Turbine has spuriously tripped.
- The Unit 2 Reactor failed to automatically scram on the Main Turbine trip.
- Manual ARI actuation has fully inserted all control rods on Unit 2.

Which one of the following identifies the highest EAL classification for these conditions?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	The highest EAL classification required for these conditions is FS1 based on loss of RCS barrier (unisolable RWCU line break) and loss of Containment barrier (unisolable primary system leakage that results in Secondary Containment area temperature > T-103 Action Level).
Distracters:	A	The higher Site Area Emergency classification is required. Plausible because MU3 does apply for the failure to scram and HU4 does apply for the seismic event > OBE.
	B	The higher Site Area Emergency classification is required. Plausible because FA1 does apply for the individual loss of the RCS barrier.
	D	A General Emergency is not required. Plausible because the RCS and Containment barriers are lost and the failure to scram could have caused fuel degradation. Additional indications for loss of Fuel Clad would result in a General Emergency, but these indications are not given.

Question 18 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1588332
User-Defined ID:	
Cross Reference Number:	204000 2.4.41
Topic:	EAL Determination - RWCU leak in RB
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	EP-AA-1007	
Learning Objective:	G6 8	
K/A System:	204000 RWCU	Importance; SRO 4.6
K/A Statement:	2.4.41 - Knowledge of the emergency action level thresholds and classifications.	
REQUIRED MATERIALS:	**Hot EAL Matrix (with Drywell radiation row blocked out on fission product barrier table)**	
Notes and Comments:	TRH 9/2/16 – The Drywell radiation row needs to be blocked out to not give away the answer for question #92).	

19 **ID: 1455731** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- Inspections have determined that four (4) of the Turbine Bypass Valves (TBVs) are inoperable and unable to perform their pressure control function on a Main Turbine trip.

Which one of the following describes how this finding affects Technical Specifications?

Technical Specification 3.7.6, Main Turbine Bypass System, LCO is...

A. still met by the remaining operable TBVs.

B. NOT met by the remaining operable TBVs. The Reactor must be shutdown.

C. NOT met by the remaining operable TBVs. Reactor operation may continue, but only if power is lowered to less than 23%.

D. NOT met by the remaining operable TBVs. Reactor operation may continue at the current power level if thermal limit penalties are satisfied.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	The COLR requires 7 operable TBVs. With 4 inoperable TBVs (of 9 total), this COLR requirement is not met. This results in TS 3.7.6 LCO not being met. The Reactor can continue to operate at the current power level as long as thermal limit penalties are satisfied. **Note:** The question meets the K/A because it presents a plant change (inoperable TBVs and resulting change in postulated Reactor pressure/power transient response, which affects reactivity) and requires candidate to use procedures (COLR, TS) to determine the effect of this plant change on continued plant operation / reactivity control.
Distracters:	A	TS 3.7.6 is not met because less than 7 TBVs are operable. Plausible because more than half of the TBVs remain operable and not all TBVs are required to be operable per the COLR.
	B	The Reactor may continue to operate at power as long as thermal limit penalties are satisfied. Plausible because a Main Turbine trip transient would be significantly more severe with 4 inoperable TBVs and TS 3.7.6 could require a significant plant down power.
	C	A down power is not required as long as thermal limit penalties are satisfied. Plausible because if thermal limit penalties cannot be satisfied, then TS 3.7.6 requires a down power to less than 23%.

Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455731
User-Defined ID:	
Cross Reference Number:	
Topic:	Inop TBVs
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(2)

Source Documentation

Source:	New Exam item **(SSES LOC27 NRC #77)** Modified Bank Bank ILT Exam Bank	**X** Previous NRC Exam Other Exam
Reference(s):	TS 3.7.6 and bases, COLR	
Learning Objective:	PLOT-2001A 14	
K/A System:		Importance; SRO 4.3
K/A Statement:	2.1.43 - Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.	
REQUIRED MATERIALS:	TS 3.2.1, TS 3.2.2, TS 3.2.3, TS 3.7.6, COLR	
Notes and Comments:		

20 **ID: 1455757** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- It is discovered that a surveillance test has not been completed on time for a Technical Specification required system.
- The surveillance frequency is 7 days.
- The surveillance was last performed 11 days ago.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the status of the system, in accordance with Technical Specifications?

A. The associated LCO must be declared NOT met at this time.

B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 2 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 7 days from the time of discovery or then the associated LCO must be declared NOT met.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	Surveillance Requirement (SR) 3.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 7 days), whichever is greater. Delay greater than 24 hours is only allowed if a risk evaluation is performed and the impact is managed. Since the risk evaluation is unable to be performed, only 24 hours are allowed in this case.
Distracters:	A	SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the surveillance has been overdue to more than 25% of the normal time requirement, such as grace period in SR 3.0.2.
	B	SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days), but only if a risk evaluation is performed. Plausible if a risk evaluation cannot be performed then 24 hours would be the correct answer.
	C	SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days). Plausible because 2 days is based on the grace period (25% of 7 days) allowed by SR 3.0.2.

Question 20 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455757
User-Defined ID:	
Cross Reference Number:	2.2.38
Topic:	Missed SR
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(1)

Source Documentation

Source:	**X** New Exam item NRC Exam **(JAF 9/14 NRC #99)** Modified Bank Bank ILT Exam Bank	**X** Previous Other Exam
Reference(s):	TS SR 3.0.3	
Learning Objective:	PLOT-1800 2	
K/A System:		Importance; SRO 4.5
K/A Statement:	2.2.38 - Knowledge of conditions and limitations in the facility license.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

21 **ID: 1140432** **Points: 1.00**

The following accident conditions exist on Unit 2:

- Drywell pressure is 8 psig and rising slowly
- Drywell oxygen is 1.2%.
- Drywell hydrogen is 1.8%.
- Torus oxygen is 1.1%.
- Torus hydrogen is 1.2%.
- Chemistry has determined that offsite release rates during Containment venting would be approximate three times (3x) ODCM limits.
- T-102, Primary Containment Control, is being executed.

Which one of the following describes the requirements for venting the Containment under these conditions in accordance with T-102?

A. Vent the Containment through a 2" vent line.

B. Vent the Containment through a 6" vent line.

C. Do NOT vent the Containment because the expected offsite release rates are too high.

D. Do NOT vent the Containment because combustible gas concentrations are below limits.

Answer: C

Answer Explanation

Choice		Basis or Justification
Correct:	C	The T-102 PC/G leg requires venting through a 2" line if Containment hydrogen concentration exceeds 0.5%, but only if the offsite release rates are expected to remain below ODCM limits during the venting evolution. Since Chemistry has determined that offsite release rates during Containment venting would be approximate three times (3x) ODCM limits, the requirement is to not vent under these conditions.
Distracters:	A	Since Chemistry has determined that offsite release rates during Containment venting would be approximate three times (3x) ODCM limits, the requirement is to not vent under these conditions. Plausible because if the expected release was lower, this would be the correct answer.
	B	Since Chemistry has determined that offsite release rates during Containment venting would be approximate three times (3x) ODCM limits, the requirement is to not vent under these conditions. Plausible because if the expected release was lower, venting would be required and a 6" vent path does exist.
	D	Containment because combustible gas concentrations have exceeded limits (0.5%). Plausible because the limit is higher on other plants and venting is not to be performed for an alternate reason.

Question 21 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1140432
User-Defined ID:	
Cross Reference Number:	2.3.11
Topic:	Containment venting due to H2
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
HIGH			10CFR55.43(b)(4)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	T-102	
Learning Objective:	PLOT-2102 7b	
K/A System:		Importance; SRO 4.3
K/A Statement:	2.3.11 – Ability to control radiation releases.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

22 **ID: 1455768** **Points: 1.00**

Unit 2 is operating at 100% power with the following:

- A loss of all Control Room annunciators occurs.
- Initial troubleshooting indicates that restoration of annunciators will take an extended period of time.

Which one of the following describes the guidance on control of Reactor power, in accordance with ON-123, Loss of Control Room Annunciators?

A Reactor power reduction is...

A. required to be performed using GP-3, Normal Plant Shutdown.

B. required to be performed using GP-9, Fast Reactor Power Reduction.

C. only to be performed if required by Technical Specifications or a Reactor scram is imminent.

D. allowed at the discretion of Shift Management using GP-9, Fast Reactor Power Reduction, but NOT required.

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	ON-123 step 8 states, IF restoration of Control Room annunciators is NOT imminent, THEN at Shift Management's discretion, **PERFORM** GP-9-2(3), "Fast Reactor Power Reduction". The given conditions state that restoration is not imminent. Therefore, execution of this step is allowed, but not required, at the discretion of the Shift Manager.
Distracters:	A	A Reactor power reduction is not required. Plausible because a Reactor power reduction is allowed under the given conditions and GP-3 is a reasonable choice for a controlled power reduction without annunciators available.
	B	A Reactor power reduction is not required. Plausible because a Reactor power reduction is allowed under the given conditions and GP-9 is the referenced procedure.
	C	A Reactor power reduction is allowed at the discretion of the Shift Manager, not just if required by TS or a scram is imminent. Plausible because maneuvering the plant without annunciators is a risk and many other plants are more restrictive in this area.

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455768
User-Defined ID:	
Cross Reference Number:	2.4.32
Topic:	ON-123 control of Reactor power
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
MEMORY			10CFR55.43(b)(5)

Source Documentation

Source:	**X** New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	ON-123	
Learning Objective:	PLOT-PBIG-1550 26a	
K/A System:		Importance; SRO 4.0
K/A Statement:	2.4.32 - Knowledge of operator response to loss of all annunciators.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

23 **ID: 1006441** **Points: 1.00**

During Refueling Operations with irradiated fuel in the RPV, water level is required to be maintained greater than or equal to 458 inches above RPV instrument zero.

Which one of the following describes the reason and applicability for this requirement, in accordance with Technical Specifications?

This requirement ensures there is sufficient water level to ___(1)___, and is applicable when moving ___(2)___.

A. (1) retain iodine fission products in the event of a fuel handling accident
(2) fuel assemblies ONLY

B. (1) retain iodine fission products in the event of a fuel handling accident
(2) fuel assemblies or handling control rods within the RPV

C. (1) limit radiation exposure to individuals performing fuel handling operations
(2) fuel assemblies ONLY

D. (1) limit radiation exposure to individuals performing fuel handling operations
(2) fuel assemblies or handling control rods within the RPV

Answer: B

Answer Explanation

	Choice		Basis or Justification
Correct:		B	(1) This is the basis as stated in Tech Spec 3.9.6 Bases. (2) This is consistent with the applicability statement of Tech Spec LCO 3.9.6.
Distracters:		A	Correct basis, incorrect applicability. Also applicable during handling of control rods. Plausible because fuel movement more directly risks fission product release based on damage to the fuel cladding.
		C	Incorrect basis, incorrect applicability. Also applicable during handling of control rods. Plausible because fuel movement more directly risks fission product release based on damage to the fuel cladding.
		D	Incorrect basis, correct applicability. Plausible because the additional water does also provide shielding for personnel working on the refuel floor.

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1006441
User-Defined ID:	13 CERT B CERT
Cross Reference Number:	G2.3.14
Topic:	ILT 1800-2-005 SRO
Num Field 1:	C CERT
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
MEMORY			10CFR55.43(b)(2)

Source Documentation

Source:	New Exam item Modified Bank Bank **X** ILT Exam Bank **(1006441)**	Previous NRC Exam Other Exam
Reference(s):	Tech Spec 3.9.6	
Learning Objective:	PLOT-1800 2	
K/A System:		Importance; SRO 3.8
K/A Statement:	2.3.14 – Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

24 **ID: 994401** **Points: 1.00**

Unit 2 is in a Refueling Outage.

In accordance with FH-6C, Core Component Movement-Core Transfers, which one of the following Refuel Floor activities MUST be DIRECTLY supervised by a Senior Reactor Operator (SRO)?

A. Cleaning recirc jet pumps in the Vessel.

B. Loading a new fuel bundle into the Vessel.

C. Removing a control rod from an empty core cell.

D. Moving LPRM strings from the core to the Spent Fuel Pool.

Answer: B

Answer Explanation		
Choice		Basis or Justification
Correct:	B	Loading a new fuel bundle into the Vessel meets the definition of a Core Alteration. All Core Alterations must be directly supervised by an SRO.
Distracters:	A	This does not require direct supervision by an SRO. Plausible because Recirc jet pumps are reactivity related components, so improper maintenance could have a negative effect on reactivity management once the plant is returned to power.
	C	This does not require direct supervision by an SRO. Plausible because moving a control rod would require direct supervision by an SRO if the control cell had one or more fuel assemblies loaded.
	D	This does not require direct supervision by an SRO. Plausible because LPRMs are a specific exception discussed in FH-6C.

Question 24 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	994401
User-Defined ID:	NLSRO-0763-2-004
Cross Reference Number:	G2.1.35
Topic:	NLSRO-0763-2-004 Unit 2 is in a Refueling Outage. Which one of the following Refuel Floor activit
Num Field 1:	0.00
Num Field 2:	0.00
Text Field:	ILT05-1 NRC Exam SRO#22

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
MEMORY			10CFR55.43(b)(6)

Source Documentation

Source:	New Exam item Previous NRC Exam Modified Bank Other Exam Bank **X** ILT Exam Bank **(994401)**	
Reference(s):	FH-6C	
Learning Objective:		
K/A System:		Importance; SRO 3.7
K/A Statement:	2.1.41 - Knowledge of the refueling process.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		

25 **ID: 1455761** **Points: 1.00**

An emergency radiological event has resulted in the following:

Time (hhmm)	Condition
0800	General Emergency (GE) declared.
0810	Notification sent with the following Protective Action Recommendations (PARs): • Evacuate 2 mile radius. • Evacuate the following sectors from 2 to 5 miles – S / SSW / SW / WSW / W.
0900	GE conditions still exist. Wind direction has changed such that the wind is from 80^0.

Based on the conditions that exist through 0900 and using Table A below, which one of the following lists the sectors that should be recommended for evacuation?

Table A			
WD (from)			Sectors
350º	to	011º	SSE / S / SSW
012º	to	034º	S / SSW /SW
035º	to	056º	SSW /SW / WSW
057º	to	079º	SW / WSW / W
080º	to	101º	WSW / W / WNW
102º	to	124º	W / WNW / NW
125º	to	146º	WNW / NW / NNW
147º	to	169º	NW / NNW / N
170º	to	191º	NNW / N / NNE
192º	to	214º	N / NNE / NE
215º	to	237º	NNE / NE / ENE
238º	to	259º	NE / ENE / E
260º	to	281º	ENE / E / ESE
282º	to	304º	E / ESE / SE
305º	to	326º	ESE / SE / SSE
327º	to	349º	SE / SSE / S

A. NNW / N/ NNE only

B. WSW / W/ WNW only

C. S / SSW / SW / WSW / W only

D. S / SSW / SW / WSW / W / WNW only

Answer: D

Answer Explanation

Choice		Basis or Justification
Correct:	D	When changes to PARs become necessary. Both the current sectors and those sectors that were evacuated in past notifications are included in the new recommendations.
Distracters:	A	Previously evacuated sectors must also be included in the new recommendation. Plausible if the candidate does not use the wind from 80 and reverses it to wind from 260.
	B	Previously evacuated sectors must also be included in the new recommendation. Plausible because current wind direction only would evacuate the WSW / W / WNW sectors.
	C	Plausible if the candidate does not understand that any new sectors must be included in the evacuation.

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	1455761
User-Defined ID:	
Cross Reference Number:	
Topic:	When PARs required
Num Field 1:	
Num Field 2:	A NRC
Text Field:	

Comments:

Psychometrics

Level of Knowledge	Difficulty	Time Allowance (minutes)	SRO
MEMORY			10CFR55.43(b)(4)

Source Documentation

Source:	X New Exam item Modified Bank ILT Exam Bank	Previous NRC Exam Other Exam Bank
Reference(s):	EP-AA-111, EP-AA-111-F-08, EP-AA-114	
Learning Objective:	G5 6	
K/A System:		Importance; SRO 4.4
K/A Statement:	2.4.29 - Knowledge of the emergency plan.	
REQUIRED MATERIALS:	**NONE**	
Notes and Comments:		