EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

1 **ID: 2087125** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor Trip
- OP-TM-EOP-001, REACTOR TRIP IMA's are complete, VSSV's are in progress
- Currently:
  - OTSG A level is 98% and rising
  - OTSG B level is 50% and lowering

What action(s) must the crew take in accordance with OP-TM-EOP-001 and what is the basis of the action?

A. (1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close
(2) Stop Feedwater Flow to the 'A' OTSG to minimize overcooling effects

B. (1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close
(2) Stop the overfeed and minimize possible water carryover or main steam line flooding

C. (1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps
(2) Stop the overfeed and minimize possible water carryover or main steam line flooding

D. (1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps
(2) Stop Feedwater Flow to the 'A' OTSG to minimize overcooling effects

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-001, REACTOR TRIP (Rev 16, Page 5) Step 3.5 directs the crew to verify both OTSG levels are less than 97.5% in the operating range; (2) if they are not then ENSURE FW-P-1A and FW-P-1B are tripped; (3) The OP-TM EOP-0011 REACTOR TRIP BASIS DOCUMENT (Rev 9, Page 10) Step 3.5 states that if feedwater is not being controlled and exceeds 97.5% then both Main Feedwater Pumps are tripped to stop the overfeed and minimize the chance of water carryover to the main steam lines.

A.	(1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close (2) Stop Feedwater Flow to the 'A' OTSG to minimize overcooling effects	INCORRECT: (1) Plausible because this will stop feedwater flow to the 'A' OTSG. Incorrect because OP-TM-EOP-001 directs tripping both Main Feedwater Pumps. (2) Plausible because this action would stop feedwater flow to the 'A' OTSG and would minimize any overcooling effects by the overfeed. Incorrect because this is not the correct action.
B.	(1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close (2) Stop the overfeed and minimize possible water carryover or main steam line flooding	INCORRECT: (1) Plausible because this will stop feedwater flow to the 'A' OTSG. Incorrect because OP-TM-EOP-001 directs tripping both Main Feedwater Pumps. (2) Correct Answer.
C.	(1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps (2) Stop the overfeed and minimize possible water carryover or main steam line flooding	CORRECT: See above.
D.	(1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps (2) Stop Feedwater Flow to the 'A' OTSG to minimize overcooling effects	INCORRECT: (1) Correct action. (2) Plausible because this would stop any overcooling effects that overfeeding may have. Incorrect because OP-TM-EOP-0011 states that tripping the main feedwater pumps at this level will prevents overfeeding into the main steam lines.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	007	K3.01	
	Importance Rating	4.0		

K/A: Reactor Trip: Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip.

Proposed Question:	Question #1	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	OP-TM-EOP-0011, Rev 9

Proposed References to be provided to applicants during examination:	None

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	EOP001-PCO-1			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the reason for a step performed in the Reactor Trip EOP.

High Cog: This question is high cog because the examinee must identify an abnormal condition and then determine which of the actions must be taken in accordance with the procedure. The examinee must correlate a high level in one OTSG to tripping both MFW pumps. The examinee must know the basis behind that action.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

2 **ID: 2096966** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Parameters indicate an increase of 0.25 gpm leakage into the RC Drain Tank
- RCS pressure steady at 2155 psig
- Ambient temperature condition at RC-RV-2, PORV, tailpipe is 100 degrees F

Based on these conditions identify the ONE selection below that describes the operation of computer alarm A0517, RC-RV-2 TAILPIPE DELTA TEMP.

If hot fluid is flowing from the PORV, A0517 will alarm at the PPC sensed **MINIMUM** temperature of ___(1)___.

A. 130° F

B. 140° F

C. 608° F

D. 618° F

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The setpoint of PPC A0517 is 30F above ambient temperature, so it will alarm at 130F tailpipe temperature. (OP-TM-PPC-A0517, Rev 4 Page 1)		
A.	130F	CORRECT: See above
B.	140F	INCORRECT: Plausible because anticipated setpoint is 40F above ambient temperature. Incorrect because the PPC takes into account a 10F bias to be conservative so the alarm comes in at 30F above ambient.
C.	608F	INCORRECT: Plausible because this is 40F below the saturation temperature of the Pressurizer. The examinee could believe the delta temperature setpoint is 40F below the Pressurizer saturation temperature as this would be closer to the actual temperature through an open PORV. Incorrect because it is 30F above ambient temperature.
D.	618F	INCORRECT: Plausible because this is 30F below the saturation temperature of the Pressurizer. The examinee could believe the delta temperature setpoint is 30F (with the 10F bias) below the Pressurizer saturation temperature as this would be closer to the actual temperature through an open PORV. Incorrect because it is 30F above ambient temperature

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008	AA2.25
	Importance Rating	3.2	

K/A: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: High-Temperature Computer alarm and alarm type

Proposed Question:	Question #2	
Technical Reference(s):	OP-TM-PPC-A0517, Rev 4	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	223-GLO-6	

Question Source:	Bank #		
	Modified Bank #	363650	
	New		
Question History:	Sim Exam 6	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have the ability to determine if the computer alarm will sense a pressurizer steam space leak pas the code safety.

High Cog: This question is high cog because the examinee must know location that the alarm based on and what the alarm setpoint is. The examinee must use math to get the correct answer.

Plant Conditions:

- Reactor is operating at 100% power with ICS in full automatic.
- Parameters indicate an increase of 0.25 gpm leakage into the RC Drain Tank.
- RCS pressure steady at 2155 psig.
- Ambient temperature condition at RC-RV-2, PORV, tailpipe is 100 degrees F.

Based on these conditions identify the ONE selection below that describes the operation of computer alarm A0517, RC-RV-2 TAILPIPE DELTA TEMP set at 30 degrees F and the minimum 19kw PZR heater margin.

If hot fluid is flowing from the PORV, A0517 will alarm at the PPC sensed **MINIMUM** temperature of ___(1)___. A steam space leak of 0.25 gpm ___(2)___ exceed the minimum 19kw PZR heater margin.

A. (1) 130° F
(2) does

B. (1) 130° F
(2) does not

C. (1) 618° F
(2) does

D. (1) 618° F
(2) does not

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

3 **ID: 2085111** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- 3 gpm RCS leak into containment
- OP-TM-AOP-050 REACTOR COOLANT LEAKAGE has been entered

Event:

- Reactor power is 88% and lowering
- RCS leak rate continues to rise
- HPI is initiated

(1) Which procedure will provide the mitigation strategy after HPI is initiated?

(2) What is the basis for using this procedure?

A. (1) Go to OP-TM-EOP-001, REACTOR TRIP
(2) To minimize the possibility of losing Subcooling Margin during a post trip cooldown

B. (1) Go to OP-TM-EOP-001, REACTOR TRIP
(2) To limit fuel centerline temperature to prevent zircaloy-water reaction

C. (1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) To locate and isolate the leak

D. (1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) To expedite shutting down the reactor

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The initial plant conditions include an RCS leak, so the crew is implementing actions in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE. OP-TM-AOP-050 (Rev 7, Page 3) Step 3.3 directs that if HPI is required to maintain pressurizer level, to initiate HPI and then trip the reactor and go to OP-TM-EOP-001, REACTOR TRIP; (2) The basis for this step in accordance with OP-TM-AOP-0501 RCS LEAKAGE BASIS DOCUMENT (Rev 7, Page 5) Step 3.3 is to minimize the possibility of losing subcooling margin during the post trip cooldown.		
A.	(1) Go to OP-TM-EOP-001, REACTOR TRIP (2) To minimize the possibility of losing Subcooling Margin during a post trip cooldown	CORRECT: See above.
B.	(1) Go to OP-TM-EOP-001, REACTOR TRIP (2) To limit fuel centerline temperature to prevent zircaloy-water reaction	INCORRECT: (1) Correct Answer. (2) Plausible because a large break loss of coolant accident could lead to a zircaloy-water reaction if equipment failed or malfunctioned and the core became uncovered. Incorrect because there is no indication in the stem to indicate that any equipment failed and/or the leak is significant enough for that reaction.
C.	(1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) To locate and isolate the leak	INCORRECT: (1) Plausible because some AOP and EOP procedures contain mitigation steps after they direct a reactor trip. Incorrect because OP-TM-AOP-050 is not one of those procedures. (2) Plausible because OP-TM-AOP-050 does contain leak isolation sections for many components.
D.	(1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) To expedite shutting down the reactor	INCORRECT: (1) Plausible because some AOP and EOP procedures contain mitigation steps after they direct a reactor trip. Incorrect because OP-TM-AOP-050 is not one of those procedures. (2) Plausible because the operators could decide to shutdown the reactor at a faster rate due to the severity of the RCS leak.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	2.4.18
	Importance Rating	3.3	

K/A: Small Break LOCA: Knowledge of specific bases for EOPs

Proposed Question:	Question #3	
Technical Reference(s):	OP-TM-AOP-050, Rev 7	OP-TM-AOP-0501, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-050-PCO-1

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #	770855	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the specific bases for an EOP entry. The question requires knowledge of the transition step bases from the Small Break LOCA AOP into the EOP network. OP-TM-EOP-001, REACTOR TRIP is entered to minimize the possibility of entering OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN. Small break LOCA at Three Mile Island is an AOP. The examinee must know when (and the basis) to initiate an EOP.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

4 **ID: 2085116** **Points: 1.00**

Due to a Large Break LOCA, OP-TM-EOP-002 LOSS OF 25F SUBCOOLING MARGIN, has been entered.

In accordance with Rule 4, FEEDWATER CONTROL, OTSG levels must be raised to __(1)__ and the basis for this level is to prepare __(2)__.

A. (1) 50% Operating Range
(2) the OTSGs for natural circulation

B. (1) 50% Operating Range
(2) for Boiler Condenser Cooling

C. (1) 75% - 85% Operating Range
(2) the OTSGs for natural circulation

D. (1) 75% - 85% Operating range
(2) for Boiler Condenser Cooling

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The crew has already entered the procedure for a loss of subcooling margin that the operators must already be performing OP-TM-EOP-010 EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS, Rule 4, FEEDWATER CONTROL (Rev 20, Page 8) as part of the OP-TM-EOP-001, REACTOR TRIP VSSV's; (2) Due to the nature of the entry into OP-TM-EOP-002, the operator must know that subcooling margin is less than 25F, and that OTSG levels must be raised to 75% to 85% in the operating range; (3) In accordance with OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT, the reason for raising levels is to promote boiler condenser cooling as a method of heat transfer to cool the reactor.

A.	(1) 50% Operating Range (2) the OTSGs for natural circulation	INCORRECT: (1) Plausible because the HSPS setpoint for this question would be 50% in the operating range. Incorrect because the examinee must raise the level to 75% to 85% in the operating range. (2) Plausible because a goal would be to restore natural circulation. Incorrect because natural circulation cannot be occurring because subcooling margin is less than 25F (OP-TM-EOP-010, Guide 10, NATURAL CIRCULATION, Rev 20, Page 23).
B.	(1) 50% Operating Range (2) for Boiler Condenser Cooling	INCORRECT: (1) Plausible because the HSPS setpoint for this question would be 50% in the operating range. Incorrect because the examinee must raise the level to 75% to 85% in the operating range. (2) Plausible because a goal would be to have boiler condenser cooling for this casualty. Incorrect because OTSG levels must be raised to higher in the operating range.
C.	(1) 75% - 85% Operating Range (2) the OTSGs for natural circulation	INCORRECT: (1) See above. (2) Plausible because a goal would be to restore natural circulation. Incorrect because natural circulation cannot be occurring because subcooling margin is less than 25F (OP-TM-EOP-010, Guide 10, NATURAL CIRCULATION, Rev 20, Page 23).
D.	(1) 75% - 85% Operating range (2) for Boiler Condenser Cooling	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011	EK1.01
	Importance Rating	4.1	

K/A: Large Break LOCA: Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling

Proposed Question:	Question #4	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	OP-TM-EOP-0101, Rev 11

Proposed References to be provided to applicants during examination:	None

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	EOP002-PCO-1				
Question Source:	Bank #				
	Modified Bank #				
	New	X			
Question History:	N/A	Last NRC Exam:	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	b.7			
	55.43				

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implication of raising OTSG level to high in the operating range. To answer the question correctly the examinee must know that their actions are to promote boiler condenser cooling during a Large Break LOCA.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

5 **ID: 2085147** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- All Reactor Coolant Pumps have tripped
- HSPS level setpoint to all four EF-V-30 valves is 0%
- Both OTSGs are 80" start-up range and lowering

Which of the following is correct regarding OTSG level control in accordance with Rule 4, FEEDWATER CONTROL?

A. Feed to 50% in the operating range with MFW to establish and maintain natural circulation

B. Throttle MFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer

C. Feed to 50% in the operating range with EFW to establish and maintain natural circulation

D. Throttle EFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Once the reactor coolant pumps have tripped, the reactor will trip; (2) This should also provide a signal for the Heat Sink Protection System to start the Emergency Feedwater Pumps and control OTSG level at 50% in the Operating Range (TQ-TM-104-644-C001, Rev 2, Page 13); (3) Once flow is lost, in addition to other things, the crew will perform OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS (Rev 20, Page 8) Rule 4, FEEDWATER CONTROL, step 4 to feed the OTSG with EFW to > 50% in the Operating Range; (6) For establishing natural circulation EFW is preferred due to the location in the OTSG that the water is introduced. EFW feeds directly on the OTSG tubes at a higher elevation than MFW.

A.	Feed to 50% in the operating range with MFW to establish and maintain natural circulation	INCORRECT: Plausible because MFW will be available to feed. Incorrect because EFW is the preferred feed source due to the location in the OTSG that the feedwater is introduced.
B.	Throttle MFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer	INCORRECT: Plausible because MFW will be available to feed. Incorrect because EFW is the preferred feed source due to the location in the OTSG that the feedwater is introduced. In addition, 25" in the SU is not the appropriate level
C.	Feed to 50% in the operating range with EFW to establish and maintain natural circulation	CORRECT: See Above
D.	Throttle EFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer	INCORRECT: Plausible because EFW is the correct feed source. Incorrect because 25" in the startup range is the wrong level.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	015		AK3.07
	Importance Rating	4.1		

K/A: Reactor Coolant Pump Malfunction: Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Ensuring that S/G levels are controlled properly for natural circulation enhancement.

Proposed Question:	Question #5	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	TQ-TM-104-644-C001, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-10

Question Source:	Bank #	2036514	
	Modified Bank #		
	New		

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the consequence of losing all Reactor Coolant pumps and the reason for the plant response. The examinee must identify a malfunction present in HSPS regarding the EFW level setpoint. The examinee must know that EFW is perferred over MFW to enhance natural circulation.

High Cog: The question is high cog because the examinee must determine the consequence of losing all Reactor Coolant pumps and have knowledge of the reason behind the plant response. The examinee must analyze the question to determine there is a malfunction with HSPS for feeding the OTSGs.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

6 **ID: 2088328** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power

Which of the following correctly identifies the correct action for restoration of makeup flow?

A. Verify MU-P-1A auto starts

B. Verify MU-P-1B auto starts

C. Manually start MU-P-1A

D. Manually start MU-P-1B

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When a Loss of Offsite Power (LOOP) occurs, all BOP and ES busses lose power, this includes the 1E 4160V bus that the normally running makeup pump (MU-P-1B) is powered from. (2) To start any makeup pump, the pump 43 selector switch must be selected to that specific pump, which also selects the pump to start on an ES. Only the pumps that are selected for ES can be started from the Control Room. (3) When power is lost, MU-P-1B will trip and will NOT be automatically restarted when the 1E 4160V bus is powered from an Emergency Diesel Generator, whereas if MU-P-1A or MU-P-1C were running when the LOOP occurred they would automatically be restarted because they are selected for ES. (4) In accordance with OP-TM-AOP-020, LOSS OF STATION POWER (Rev 24, Page 3) on Step 3.6 the operators will verify Seal Injection flow is greater than 22 GPM, which since no Makeup Pump is running, it will be 0 gpm. (5) The operators will initiate OP-TM-AOP-041, LOSS OF SEAL INJECTION (Rev 8, Page 3) will lead the operators to starting MU-P-1A (Step 3.5 RNO step 5) on its current power supply, the 1D 4160V bus.		
A.	Verify MU-P-1A auto starts	INCORRECT: Plausible because MU-P-1A will be started on the 1D 4160V bus. Incorrect because it will not auto start.
B.	Verify MU-P-1B auto starts	INCORRECT: Plausible because if MU-P-1B were ES selected (which it normally is NOT) it would have auto started on the 1E 4160V bus. Incorrect because it is not ES selected.
C.	Manually start MU-P-1A	CORRECT: See above.
D.	Manually start MU-P-1B	INCORRECT: Plausible because OP-TM-AOP-041 does have a section to manually start MU-P-1B on the 1E 4160V bus. Incorrect because that section would only be performed if MU-P-1A could not be started.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AA2.02
	Importance Rating	3.2	

K/A: Loss of Reactor Coolant Makeup: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup Pumps: Charging pump problems

Proposed Question:	Question #6	
Technical Reference(s):	OP-TM-AOP-041, Rev 8	OP-TM-AOP-020, Rev 24

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-020-PCO-4	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know that a loss of reactor coolant makeup occurs during a loss of offsite power in this lineup. The examinee must know the correct makeup pump to start. A loss of makeup pump power is a charging pump problem.

High Cog: The examinee must understand the different power supplies for the makeup pumps and recognize the conditions required for a Makeup Pump to auto start.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

7 **ID: 2088354** **Points: 1.00**

Plant conditions:

- Plant is in Refueling Shutdown
- Decay Heat Removal (DHR) Train A is in service

Event:

- DC-T-1A Decay Heat Surge Tank level is rising slowly
- Decay Heat Leak is estimated at ~ 2gpm

Which of the following identifies the correct action the crew must take and the procedure that directs the action?

The crew must __(1)__ in accordance with __(2)__.

A. (1) trip Decay Heat Removal Pump, DH-P-1A immediately
(2) OP-TM-AOP-060, LEAKAGE WHILE ON DECAY HEAT REMOVAL

B. (1) trip Decay Heat Removal Pump, DH-P-1A immediately
(2) OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL

C. (1) place DHR train B in service
(2) OP-TM-AOP-060 LEAKAGE WHILE ON DECAY HEAT REMOVAL

D. (1) place DHR train B in service
(2) OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) During refueling operations, the normal lineup is to have one string of the Decay Heat Removal (DHR) system in operation with the other string in standby. (2) When the leak begins, the entry criteria to OP-TM-AOP-060, LEAKAGE WHILE ON DECAY HEAT REMOVAL (Rev 8, Page 1) is met. (3) The crew would identify the leak in the operating DHR string into the Decay Closed System which is in the Auxiliary Building. When the crew reached Step 4.6 (Rev 8, Page 19) DHR string B would be put into service.		
A.	(1) trip Decay Heat Removal Pump, DH-P-1A immediately (2) OP-TM-AOP-060, LEAKAGE WHILE ON DECAY HEAT REMOVAL	INCORRECT: (1) Plausible because an early step in OP-TM-AOP-060 would require tripping DH-P-1A if signs of vortexing were present. Vortexing could happen if the Fuel Transfer Canal level lowered too much. Incorrect because no indication that vortexing is occurring. (2) Correct Answer.
B.	(1) trip Decay Heat Removal Pump, DH-P-1A immediately (2) OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL	INCORRECT: (1) Plausible because an early step in OP-TM-AOP-060 would require tripping DH-P-1A if signs of vortexing were present. Vortexing could happen if the Fuel Transfer Canal level lowered too much. Incorrect because no indication that vortexing exists. (2) Plausible if the examinee believes that the entry criteria are met to enter this EOP. Currently one train has a leak and the other is not currently operating. Incorrect because the DHR B is available.
C.	(1) place DHR train B in service (2) OP-TM-AOP-060 LEAKAGE WHILE ON DECAY HEAT REMOVAL	CORRECT: See above.
D.	(1) place DHR train B in service (2) OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that the entry criteria are met to enter this EOP for both DHR trains being unavailable. Currently one train has a leak and the other is not currently operating. Incorrect because the DHR B is available.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AA2.02
	Importance Rating	4.1	

K/A: Loss of Residual Heat Removal: Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal: Shift to alternate flowpath

Proposed Question:	Question #7	
Technical Reference(s):	OP-TM-AOP-060, Rev 8	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-060-PCO-2	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the reason to change the residual heat removal flowpath. The examinee must have knowledge that a leak in the operating DHR string will required starting the standby string.

High Cog: This question is high cog because the examinee must determine that starting the B DHR string is required and that immediately tripping DH-P-1A is not required. In addition, they must recognize the entry criteria into the correct procedure.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

8 **ID: 2088412** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Main Instrument Air Compressor, IA-P-4 has been secured for maintenance
- Instrument Air compressors IA-P-1A and IA-P-1B are operating

Event:

- Loss of Offsite Power

10 Minutes Later:

- Due to low instrument air pressure, the CRO re-starts IA-P-1A
- PLB-7-4 EFP Room A Sump Level High Alarm actuates

Which of the following describes the current cooling to IA-P-1A and parameter/components that actuated the interlock?

Cooling to IA-P-1A is aligned to __(1)__ due to __(2)__.

A. (1) Domestic Water
(2) Secondary Closed Cooling Water flow

B. (1) Domestic Water
(2) Secondary Closed Cooling Water pump breaker position

C. (1) Fire Service Water
(2) Secondary Closed Cooling Water flow

D. (1) Fire Service Water
(2) Secondary Closed Cooling Water pump breaker position

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The normal cooling water to the IA-P-1's is Secondary Closed Cooling Water (TQ-TM-104-850-C001, Rev 6, Page 9). (2) On a Loss of Offsite Power the breakers for all Secondary Closed Cooling Water pumps open on interlock. (3) The new cooling to the IA-P-1's is Fire Service Water. (4) Alarm PLB-7-4 comes in because the Fire Service Water flows through the compressor cooler to the Intermediate Building Sump.

A.	(1) Domestic Water (2) Secondary Closed Cooling Water flow	INCORRECT: (1) Plausible if the examinee does not know the backup cooling to the IA-P-1's. Incorrect because Fire Service is the backup. (2) Plausible because the examinee could believe the interlock actuates of the Secondary Closed Cooling Water flow.
B.	(1) Domestic Water (2) Secondary Closed Cooling Water pump breaker positions	INCORRECT: (1) Plausible if the examinee does not know the backup cooling to the IA-P-1's. Incorrect because Fire Service is the backup. (2) Correct answer.
C.	(1) Fire Service Water (2) Secondary Closed Cooling Water flow	INCORRECT: (1) Correct answer. (2) Plausible because the examinee could believe the interlock actuates of the Secondary Closed Cooling Water flow.
D.	(1) Fire Service Water (2) Secondary Closed Cooling Water pump breaker positions	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	026	AA1.03	
	Importance Rating	3.6		

K/A: Loss of Component Cooling Water (CCW): Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water: SWS as a backup to CCWS

Proposed Question:	Question #8	
Technical Reference(s):	TQ-TM-850-C001, Rev 6	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	851-GLO-10	

Question Source:	Bank #		
	Modified Bank #	1718269	
	New		

Question History:	N/A	Last NRC Exam:	N/A	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

Modified: The unmodified question was in the ILT 16-01 NRC Exam.

KA Match: This question matches the KA because the examinee must know that Fire Service water (SWS) is the back up to Secondary Closed Cooling Water. The Secondary Closed Cooling Water system is lost when the LOOP occurs.

High Cog: This question is high cog because the examinee must know analyze the stem and determine how the LOOP affects the IA-P-1s.

MODIFIED FROM:

Plant Conditions:
- 100% power with ICS in full auto.
- Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
- Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

EVENTS:

- LOCA
- Loss of Off-Site Power
- 1600 PSI ES Actuation

10 minutes later:

- Due to low instrument air pressure, the CRO starts IA-P-1A.

Cooling for IA-P-1A/B will be aligned to __(1)__ due to the __(2)__.

A. (1) Fire Service Water, (2) ES Actuation Signal.

B. (1) Fire Service Water, (2) Loss of Offsite Power.

C. (1) Secondary Closed Cooling, (2) ES Actuation Signal.

D. (1) Secondary Closed Cooling, (2) Loss of Offsite Power.

Answer: B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

9 **ID: 2088420** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Annunciator "ICS/NNI POWER LOST" H-1-8 is received
- ICS AUTO Power (30 amp fuse) has been lost
- Pressurizer pressure is 2145 psig and lowering
- Pressurizer level, as indicated on LI-777A, is 220 inches and relatively steady

Which one of the following identifies the action(s) that must be taken?

A. Bypass Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 in Auto

B. Energize Pressurizer Heater Banks 4 & 5 using the "ON/OFF" switches on CR until ICS Auto Power is restored

C. Bypass the Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 from the Bailey Station on CC

D. Take manual control of FW-V-16A/B & 17A/B using the HAND/AUTO Stations on CC and close as necessary to prevent the excessive RCS cooldown

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When MAP H-1-8 alarms, in conjunction with the knowledge that the 30 amp ICS Auto light is blown, the crew must enter OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER (Rev 10). (2) With ICS Auto power lost, the crew must place the PZR HTR LO-LO LEVEL CUTOUT BYPASS switch to BYPASS (OP-TM-AOP-027, Rev 5 Page 5) in step 3.13, and initiate OP-TM-220-503, MANUAL CONTROL OF PRESSURIZER PRESSURE in step 3.15 (OP-TM-AOP-027, Rev 5, Page 7). (3) The operator must know that with ICS Hand power still available that heater groups 1 through 3 are operated in with the toggle switch on the Bailey Station (OP-TM-220-503, Step 4.1, Rev 4, Page 2)

A.	Bypass Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 in Auto	INCORRECT: Plausible because the Pressurizer Lo-Lo Level Interlock must be bypassed. Incorrect because the Pressurizer heaters are still not operable in automatic at this point.
B.	Energize Pressurizer Heater Banks 4 & 5 using the "ON/OFF" switches on CR until ICS Auto Power is restored	INCORRECT: Plausible if the examinee believes the ON/OFF function of the Group 4 and 5 heaters is still available. Incorrect because the Pressurizer Lo-Lo Level Interlock must be bypassed to operate those heaters.
C.	Bypass the Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 from the Bailey Station on CC	CORRECT: See above.
D.	Take manual control of FW-V-16A/B & 17A/B using the HAND/AUTO Stations on CC and close as necessary to prevent the excessive RCS cooldown	INCORRECT: Plausible since OP-TM-AOP-027 directs manual control or FW-V-16A/B and 17A/B. Incorrect because at 100% power an excessive RCS cooldown will not occur and these FW valves will not need to be operated.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	027	AK2.03	
	Importance Rating	2.6		

K/A: Pressurizer Pressure Control System Malfunction: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and Positioners

Proposed Question:	Question #9	
Technical Reference(s):	OP-AOP-027, Rev 10	OP-TM-220-503, Rev 4
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	624-GLO-9	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #	299478	
	Modified Bank #		
	New		

Question History:	Unmod on Sim Exam 7	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know that the loss of ICS Auto power malfunction causes the pressurizer heaters to shutoff due to loss of level signal. The operator must operate the bypass feature to regain control of the pressurizer heater banks controller.

High Cog: The examinee will have to analyze the given conditions to determine the procedure path. The examinee must also understand how the loss of ICS Auto Power impacts Pressurizer Heater Control for both the SCR and analog controlled heaters.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

10 **ID: 2085315** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- Reactor power is 50% with ICS in auto

Sequence of Events:

- ICS placed in HAND due to a feedwater transient that caused RCS temperature to rise
- An RCS leak occurs
- Currently:
    - RCS pressure is 1950 psig
    - RCS Thot is 617°F

Based on the above information, an ATWS has:

A. NOT occurred and Main Feedwater flow must be raised

B. NOT occurred and the reactor must be tripped if the RCS leak cannot be isolated

C. occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and HPI initiation is required if reactor power remains at 50%

D. occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and a manual turbine trip is required if reactor power remains at 50%

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Using the provided reference, Technical Specification Figure 2.3-1, that an ATWS has occurred due to exceeding the Variable Low Pressure Trip. For most of Figure 2.3-1, 618.8F is the highest temperature allowed and 1900 psig is the highest pressure allowed. However, the Variable Low Pressure Trip (VLPT) is in effect on the bottom right hand corner of the figure. (2) OP-TM-EOP-001, REACTOR TRIP (Rev 16, Page 1) must be entered and the IMA's performed. (3) Based the nature of the correct answer and the other ATWS distractor, it is implied that pressing the Reactor Trip and DSS pushbuttons fail since both mention 'if reactor power stays at 50%'. (4) The RNO of step 2.2 is entered and the and because the Reactor is still at 50% when RNO step 5 is reached, HPI is then initiated.		
A.	NOT occurred and Main Feedwater flow must be raised	INCORRECT: Plausible if the examinee does not correctly take into consideration the VLPT. In addition, the crew would slowly raise Main Feedwater flow to lower the temperature while dealing with the RCS leak. Incorrect because the VLPT was exceeded.
B.	NOT occurred and the reactor must be tripped if the RCS leak cannot be isolated	INCORRECT: Plausible if the examinee does not correctly take into consideration the VLPT. Because the examinee does not believe an ATWS occurred, they will continue in OP-TM-AOP-050, REACTOR COOLANT LEAK, which will continue with a plant shutdown. The reactor would be tripped if an RCS trip setpoint is approached or exceeded. Incorrect because the VLPT was exceeded.
C.	occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and HPI initiation is required if reactor power remains at 50%	CORRECT: See above.
D.	occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and a manual turbine trip is required if reactor power remains at 50%	INCORRECT: Plausible because the third step of the RNO section could lead the examinee to believe that tripping the Turbine is the correct answer. Incorrect because the Turbine would only be tripped if Main Feedwater was not available.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029	EK3.02
	Importance Rating	3.1	

K/A: Anticipated Transient Without Scram: Knowledge of the reasons for the following responses as they apply to an ATWS: Starting a specific charging pump

Proposed Question:	Question #10	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	Tech Spec Figure 2.3-1, AMD 262

EXAMINATION ANSWER KEY

Proposed References to be provided to applicants during examination:			Fig 2.3-1, AMD 262
Learning Objective:	641-GLO-10		
Question Source:	Bank #	978945	
	Modified Bank #		
	New		
Question History:	N/A	Last NRC Exam:	12-01

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the reason the charging pumps are started in order to answer the question correctly. The examinee must identify there is an ATWS, then know that the charging pumps are started because the reactor did not shutdown (vice tripping the turbine). At Three Mile Island there is no reason a specific charging pump would be started for an ATWS. This question starts all of the charging pumps.

High Cog: The question is at the Comprehension/Analysis cognitive level because candidate must know the operation curves of Tech Spec Figure 2.3-1 and understand that the variable high temperature and pressure should cause a reactor trip.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

11 **ID: 2085318** **Points: 1.00**

Plant Conditions:

- A tube rupture has occurred in the "A" OTSG
- The reactor has been tripped
- Current 5-minute RCS cooldown rate is 100°F/ hr
- "A" loop RCS cold leg temperature is 540°F
- "B" loop RCS cold leg temperature is 543°F
- "A" OTSG Turbine Bypass Valves, MS-V-3D/E/F, are open 30% in manual
- "B" OTSG Turbine Bypass Valves, MS-V-3A/B/C, are closed
- RCS pressure is 1210 psig
- "A" OTSG level is 82% and rising at 3% / minute

Which of the following actions must the crew take to mitigate this casualty?

A. Immediately lower RCS pressure to < 1000 psig using the PORV, RC-RV-2

B. Immediately lower RCS pressure to < 1000 psig using the Spray Valve, RC-V-1

C. Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3A/B/C

D. Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3D/E/F

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Due to the 'A' OTSG tube rupture, the crew is in OP-TM-EOP-005, OTSG TUBE LEAKAGE (Rev 10). The reactor is tripped and a cooldown is commenced. Due to the "A" OTSG Turbine Bypass Valve position, it is evident that the crew is preferentially steaming the "A" OTSG to cooldown the plant at the 100F/hr limit. (2) At 85% in the operating range (Step 3.34, Page 17) the crew must isolate the "A" OTSG, but RCS pressure must be below 1000 psig. (3) Due to challenging isolation criteria, in accordance with Step 3.31 (Page 15), the crew must initiate an RCS cooldown to ≤ 500F Thot at a rate < 240F/hr using the "A" OTSG Turbine Bypass Valves. (4) Using the "A" OTSG Turbine Bypass Valves will allow for the plant to be cooldown and the "A" OTSG level to lower (or not rise as fast) to maximize time before the isolation criteria is met.		
A.	Immediately lower RCS pressure to < 1000 psig using the PORV, RC-RV-2	INCORRECT: Plausible because in Step 3.31 the crew will open the PORV AFTER initiating the RCS cooldown. Incorrect because the crew does not immediately perform this step.
B.	Immediately lower RCS pressure to < 1000 psig using the Spray Valve, RC-V-1	INCORRECT: Plausible because in Step 3.31 the crew will open the Spray Valve AFTER initiating the RCS cooldown. Incorrect because the crew does not immediately perform this step.
C.	Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3A/B/C	INCORRECT: Plausible because the examinee must maximize cooldown rate. Incorrect because the crew must preferentially steam the "A" OTSG, not the "B" OTSG.
D.	Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3D/E/F	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038	EA1.36
	Importance Rating	4.3	

K/A: Steam Generator Tube Rupture: Ability to operate and monitor the following as they apply to a SGTR: Cooldown to a specified temperature

Proposed Question:	Question #11	
Technical Reference(s):	OP-TM-EOP-005, Rev 10	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EOP005-PCO-5	

Question Source:	Bank #	862872	
	Modified Bank #		
	New		

Question History:	Simulator Exam 9	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the cooldown rate to a specific temperature for an OTSG tube rupture.

High Cog: This question is high cog because the examinee analyzes the parameters (the act of preferentially steaming and OTSG isolation criteria) and know the procedure steps that apply. The examinee must identify the correct action to take.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

12 **ID: 2087712** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

A Steam Line Rupture occurs, the crew trips the reactor and enters OP-TM-EOP-003, EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER.

Reactor Building pressure peaks at 12 psig.

Which phases of isolation must be complete and where must pressure/temperature be stabilized in accordance with EOP-003?

A. (1) Phase 1 only
(2) The RCS Pressure/Temperature after the XHT is terminated

B. (1) Phase 1 Only
(2) At Hot Shutdown Values

C. (1) Phase 1 and 2
(2) The RCS Pressure/Temperature after the XHT is terminated

D. (1) Phase 1 and 2
(2) At Hot Shutdown Values

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The first immediate manual action of EOP-003 (Rev 9, Page 1) is to perform Rule 3, EXCESSIVE HEAT TRANSFER of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS (Rev 20, page 6). Rule 3 directs the operator to perform Phase 1 isolation (Steam outputs of the OTSG), then verify the leak is not in the Reactor or Intermediate Buildings. (2) Reactor Building pressure peaking at 12 psig indicates that the leak is in the Reactor Building. (3) This requires the operator to go to the RNO and perform Phase 2 isolation (EFW cooling to the OTSG). (4) When Phase 1 and 2 isolations are complete, there is no feed to the OTSG. After the remaining steam is released from the OTSG the cooldown will be terminated. (5) EOP-003 (Rev 9, Page 3) directs the crew to perform Guide 12, RCS STABILIZATION of EOP-010 (Rev 20, Page 25), which directs the crew to adjust the remaining OTSG Pressure to stabilize RCS temperature and throttle HPI at the current values post cooldown (pressure/temperature when the XHT is terminated). (6) The basis for Guide 12 in OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT (Rev 11, Pages 50 and 51) is to stabilize plant (pressure and temperature) after the cooldown has subsided to prevent further stresses and pressurized thermal shock.

A.	(1) Phase 1 only (2) The RCS Pressure/Temperature after the XHT is terminated	INCORRECT: (1) Plausible because the question does not explicitly state that the Steam Leak is in the Reactor Building. Incorrect because it is unisolable in the Reactor Building based on the elevated Reactor building pressure. (2) The RCS Pressure/Temperature after the XHT is terminated is the correct answer.
B.	(1) Phase 1 Only (2) At Hot Shutdown Values	INCORRECT: (1) Plausible because the question does not explicitly state that the Steam Leak is in the Reactor Building. Incorrect because it is unisolable in the Reactor Building based on the elevated Reactor building pressure. (2) Plausible because the Hot Shutdown values are where the operators normally stabilize the plant after a reactor trip. Incorrect because the goal after an overcooling event is to keep temperature and pressure stable at the lowest value possible to minimize further stress on OTSG and Reactor components.
C.	(1) Phase 1 and 2 (2) The RCS Pressure/Temperature after the XHT is terminated	CORRECT: See above.
D.	(1) Phase 1 and 2 (2) At Hot Shutdown Values	INCORRECT: (1) Phase 1 and 2 is the correct answer. (2) Plausible because the Hot Shutdown values are where the operators normally stabilize the plant after a reactor trip. Incorrect because the goal after an overcooling event is to keep temperature and pressure stable at the lowest value possible to minimize further stress on OTSG and Reactor components.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	040	AK1.04	
	Importance Rating	3.2		

K/A: Steam Line Rupture - Excessive Heat Transfer: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Nil Ductility Temperature

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Proposed Question:	Question #12		
Technical Reference(s):	OP-TM-EOP-003, Rev 9	OP-TM-EOP-010, Rev 20	
	OP-TM-EOP-001, Rev 11		
Proposed References to be provided to applicants during examination:		None	
Learning Objective:	EOP003-PCO-4		
Question Source:	Bank #		
	Modified Bank #		
	New	X	
Question History:	N/A	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the K/A because the examinee must know how to operationally control excessive RCS heat transfer. When the plant uncontrollably overcools the possibility of components reaching their Nil-ductility temperature rises, so when the cooldown has subsided the operators should maintain the RCS temperature/pressure at the post cooldown values.

High Cog: This question is high cog because the examinee must analze the stem and determine that Phase 2 isolation needs to be completed. In addition, the examinee must know where to stabilize the RCS parameters.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

13 **ID: 2096952** **Points: 1.00**

Plant conditions:

- Plant startup in progress
- Reactor Power is 12%
- Both Aux Boilers are secured
- Chest Warming is in progress on the main turbine
- FW-P-1A is on line
- FW-P-1B is secured

Events:

- A large steam leak occurs in the Intermediate Building
- "A" OTSG pressure lowers to 500 psig
- The CRO trips the reactor and performs Phase 1 isolation of "A" OTSG, only

Assuming no further operator actions, what is the current response of Main and Emergency Feedwater flow to (1) "A" and (2) "B" OTSG's?

A. (1) "A" OTSG Main and Emergency Feedwater are isolated by HSPS
(2) "B" OTSG Main Feedwater will maintain level at the 25" setpoint

B. (1) "A" OTSG Main and Emergency Feedwater are isolated by HSPS
(2) "B" OTSG Main Feedwater flow will fall to zero lbm/hr and Emergency Feedwater will maintain at the 25" setpoint

C. (1) "A" OTSG Main Feedwater ONLY is isolated by HSPS and Emergency Feedwater will be feeding "A" OTSG
(2) "B" OTSG Main Feedwater will maintain level at the 25" setpoint

D. (1) "A" OTSG Main Feedwater ONLY is isolated by HSPS and Emergency Feedwater will be feeding "A" OTSG
(2) "B" OTSG Main Feedwater flow will fall to zero lbm/hr and Emergency Feedwater will maintain at the 25" setpoint

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The Heat Sink Protection System (HSPS) will isolate Main Feedwater to the "A" OTSG when pressure lowers to less than 600 psig and the system is not bypassed. (TQ-TM-104-644-C001, Rev 2 Page 29). (2) HSPS will also initiate EFW on Low OTSG Level, which will occur in any OTSG after level reaches 10" in the startup range. (TQ-TM-104-644-C001, Rev 2 Page 30). (3) When MS-V-1A and MS-V-1B (during Phase 1 Isolation of the OTSG) are closed, that removes the only available source of steam to FW-P-1A, which secures all available MFW sources (i.e. all MFW flow drops to 0 lbm/hr). (4) Since Phase 2 isolation was not performed on the "A" OTSG (OP-TM-EOP-010, Rule 3, Rev 10 pages 6 and 7) then EFW is still available to feed the OTSG. (5) When OTSG level lowers to "B" OTSG, then EFW will be initiated and feed to keep level at 25" in the startup range.		
A.	1) "A" OTSG Main and Emergency Feedwater are isolated by HSPS (2) "B" OTSG Main Feedwater will maintain level at the 25" setpoint	INCORRECT: (1) Plausible if the examinee believes HSPS will isolate MFW and EFW. Incorrect because HSPS only isolates MFW. (2) Plausible of the examinee believes that MS-V-1A and MS-V-1B isolate FW-P-1B. The level setpoint would be 25" if MFW were feeding the OTSG. Incorrect because those valves isolate FW-P-1A, which is the only running Main Feedwater Pump.
B.	(1) "A" OTSG Main and Emergency Feedwater are isolated by HSPS (2) "B" OTSG Main Feedwater flow will fall to zero lbm/hr and Emergency Feedwater will maintain at the 25" setpoint	INCORRECT: (1) Plausible if the examinee believes HSPS will isolate MFW and EFW. Incorrect because HSPS only isolates MFW. (2) Correct answer.
C.	(1) "A" OTSG Main Feedwater ONLY is isolated by HSPS and Emergency Feedwater will be feeding "A" OTSG (2) "B" OTSG Main Feedwater will maintain level at the 25" setpoint	INCORRECT: (1) Correct answer. (2) Plausible of the examinee believes that MS-V-1A and MS-V-1B isolate FW-P-1B. The level setpoint would be 25" if MFW were feeding the OTSG. Incorrect because those valves isolate FW-P-1A, which is the only running Main Feedwater Pump.
D.	(1) "A" OTSG Main Feedwater ONLY is isolated by HSPS and Emergency Feedwater will be feeding "A" OTSG (2) "B" OTSG Main Feedwater flow will fall to zero lbm/hr and Emergency Feedwater will maintain at the 25" setpoint	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	054	AA2.02	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

		Importance Rating	4.1		

K/A: Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Differentiation between loss of all MFW and trip of one MFW pump

Proposed Question:	Question #13	
Technical Reference(s):	TQ-TM-104-644-C001, Rev 2	OP-TM-EOP-010, Rev 20

Proposed References to be provided to applicants during examination:	NONE

Learning Objective:	401-GLO-11	

Question Source:	Bank #	689340	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	
	55.43	

Comments:

KA Match: This question matches the KA because the examinee will have to differentiate between a loss of one main feedwater pump and all main feedwater pumps. The examinee will have to determine that the isolation valves to the only running main feedwater pump are closed.

High Cog: This question is high cog because the examinee must analyze the plant response when Phase 1 isolation is complete. In addition, the examinee must identify that the "A" OTSG pressure is below 600 psig and that HSPS isolated MFW to that OTSG.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

14 **ID: 2096960** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power

Current Parameters:

- RCS SCM is 70°F
- OTSG 1A/1B pressures are 850 psig
- OTSG 1A/1B levels are 45% and rising at 5%/minute
- Emergency Feedwater valves EF-V-30A-D are 100% open in automatic control
- RCS natural circulation cooling has been verified
- Atmospheric dump valves (ADVs) MS-V-4A/B are both closed in automatic

Identify the selection below that describes the resulting operational implications on natural circulation flow after the OTSGs have reached their level setpoint.

Natural circulation flow rate will _______.

A. rise because OTSG temperature rises.

B. not be affected because OTSGs are at level setpoint.

C. initially lower as RCS temperature rises and then stabilize at a lower value.

D. not be affected as long as steam is being supplied to the Steam Driven EFW Pump, EF-P-1.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) A differential temperature develops proportional to decay heat and heat. B & W plant experience has shown that the temperature difference can approach 50F following reactor trip and loss of forced flow. As decay heat lowers, this temperature difference lowers proportionally. If the differential temperature continues to rise that would be an indication either lack of heat transfer or lack of heat sink. (OP-TM-EOP-0101, Rev 11 Page 45) (2) At the point in which the question asks the status of the natural circulation, EFW flow (currently the only Feedwater available) stops flowing into the OTSG. This would lower the differential temperature on the RCS in that FW is not providing as much cooling for the OTSG tubes. (3) The steam indicates currently the ADVs are closed, so at the point at which EFW stops feeding and the ADV's are closed natural circulation will initially lower and stabilize at a lower value. (OP-TM-EOP-010, Rev 20 Page 41)

A.	rise because OTSG temperature rises	INCORRECT: Plausible because OTSG temperature will rise. Incorrect because the temperature rise would be in the cold leg, which reduce flow due to lower difference in RCS density.
B.	not be affected because OTSGs are at level setpoint	INCORRECT: Plausible because natural circulation is also effected by steaming the OTSG's (i.e. opening the ADVs0. Incorrect because feeding the OTSGs with EFW also promotes natural circulation.
C.	initially reduce as RCS temperature rises and then stabilize at a lower value	CORRECT: See above.
D.	not be affected as long as steam is being supplied to the Steam Driven EFW Pump, EF-P-1	INCORRECT: Plausible because the examinee could believe that as long as steam is being drawn off for EF-P-1 that natural circulation will not change (i.e. steaming the OTSG's). Incorrect because the question is strictly asking about natural circulation when the EFW setpoint is reached. Steam to EF-P-1 is negligible at that point.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	056	AK1.01	
	Importance Rating	3.7		

K/A: Loss of Offsite Power: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural circulation

Proposed Question:	Question #14	
Technical Reference(s):	OP-TM-EOP-0101, Rev 11	OP-TM-EOP-010, Rev 20

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-020-PCO-5	

Question Source:	Bank #	363884	
	Modified Bank #		

	New		
Question History:	ILT 18-01 Sim Exam 8	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.14
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implication that reaching the OTSG level setpoint in automatic will have on natural circulation.

High Cog: The examinee has to analyze the conditions given in the question stem and determine that since the OTSG pressure is lower than the ADV setpoint, that when EFW reaches its level setpoint the amount of natural circulation will lower.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

15 **ID: 2085756** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- MAP A-1-6, INVERTER FAILED illuminates
- MAP A-3-8, INVERTER 1B/1D/1F TROUBLE illuminates
- D RPS cabinet de energized (no lights)
- ES Train B Channel trip indication on Panel PCR

What action must the crew take and what is the basis for this action?

A. PLACE RC-P-2B-2 in PTL to reduce load on the "B" DC system

B. PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation

C. PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of Aux and FH building ventilation

D. PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-A position to align the ES status light indications to the "A" Vital Bus to restore status indications

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must: (1) Analyze the conditions and determine that a loss of a loss of Vital Bus has occurred (the definitive bullets being the "D" RPS cabinet de-energized, and the ES train "B" Channel Trip). (2) OP-TM-AOP-018, LOSS OF VBD (Rev 9 Page 3) directs the crew to defeat RM-A-1G and restore Control Building Ventilation.		
A.	PLACE RC-P-2B-2 in PTL to reduce load on the "B" DC system	INCORRECT: Plausible if students believe a loss of DC "B" occurred based on plant indications.
B.	PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation	CORRECT: See above.
C.	PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of Aux and FH building ventilation	INCORRECT: Plausible because these actions would apply if Vital Bus "C" is lost. Incorrect because the loss of Vital Bus "D" has occurred.
D.	PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-A position to align the ES status light indications to the "A" Vital Bus to restore status indications	INCORRECT: Plausible because these actions would apply if Vital Bus "B" is lost. Incorrect because the loss of Vital Bus "D" has occurred.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	057	AA2.20	
	Importance Rating	3.6		

K/A: Loss of Vital AC Instrument Bus: Ability to determine and interpret the following to the Loss of Vital AC Instrument Bus: Interlocks in effect on loss of ac vital electrical instrument bus that must be bypassed to restore normal equipment operation

Proposed Question:	Question #15	
Technical Reference(s):	OP-TM-AOP-018, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP18-PCO-5	

Question Source:	Bank #		
	Modified Bank #	1720293	
	New		

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must determine the Vital Bus that was lost and identify the component which must be defeated to restore control building ventilation.

High Cog: Examinee must analyze plant conditions and indications to identify the Vital Bus that was loss and the proper actions to take to restore equipment operation of equipment.

Plant Event

- Numerous alarms occur simultaneously in the control room while operating at 100% power. The following indications are noted by the crew:
  - MAP A-1-6, INVERTER FAILED illuminates.
  - MAP A-3-8, INVERTER 1B/1D/1F TROUBLE illuminates.

Which ONE of the following describes the required action per procedure AND the reason?

A. PLACE RC-P-2B-2 in PTL to reduce load on the "B" DC system.

B. PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation.

C. PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of AB and FHB ventilation.

D. PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-B position to align the ES status light indications to the "B" Vital Bus to restore status indications.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

16 **ID: 2085769** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto
- EG-Y-1B is OOS for Maintenance
- NR-P-1B is powered by the 1R 480V ES Bus and selected for ES in accordance with OP-TM-541-443, SWAP NR-P-1B TO ALTERNATE POWER SUPPLY
- NR-P-1A and NR-P-1C are running

Sequence of Events:

- Loss of Offsite Power
- Large Break LOCA occurs 30 seconds later
- 4 psig ES actuation

Which of the following is the correct response for these events?

A. NR-P-1A will auto start

B. NR-P-1B will be off

C. NR-V-1A will be closing

D. NR-V-1B will be opening

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) For this stem, when offsite power is lost, the Nuclear River Water pump in standby will start. (2) When a Nuclear River Water pump starts, its outlet valve (NR-P-1B) will travel open. (3) The valve only goes closed then the manual CLOSE pushbutton is pushed (TQ-TM-104-531-C001, PRIMARY COOLING SYSTEMS (Rev 9 Pages 9 & 37) a. NR-V-1A/B/C automatically open on pump start signal and MUST be manually closed on pump stop. (4) The non-ES selected pump (NR-P-1A) will trip on an ES actuation.

A.	NR-P-1A will auto start	INCORRECT: Plausible because in a normal plant lineup, NR-P-1A will auto start when the ES actuated. Incorrect because NR-P-1B is ES selected on the 1R 480V ES bus.
B.	NR-P-1B will be off	INCORRECT: Plausible because in a normal plant lineup NR-P-1B will have started when the LOOP occurred, then tripped when the ES occurred. Incorrect because NR-P-1B is ES selected on the 1R 480V ES bus.
C.	NR-V-1A will be closing	INCORRECT: Plausible because some ES valves (Decay River) valves go closed when the pump trips. Incorrect because the NR-V-1's do not have that interlock.
D.	NR-V-1B will be opening	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	062	AK3.01	
	Importance Rating	3.2		

K/A: Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the Nuclear Service Water Coolers.

Proposed Question:	Question 16	
Technical Reference(s):	TQ-TM-104-531-C001, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	531-GLO-5

Question Source:	Bank #	862968
	Modified Bank #	
	New	

Question History:	System Exam 6	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the opening signal to the river water isolation valve to the Nuclear Service Heat Exchangers. These are the only isolation valves to in the Nuclear Service Water System that isolate part of the heat exchanger that have automatic features. These valves to not have an automatic closing water feature.

High Cog: Examinee has to analyze the plant conditions, the power supply switch and the effect of the LOOP with an ES actuation signal has on the effected components.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

17 **ID: 2085775** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Annunciator PLB-1-7 INSTRUMENT AIR PRESS LOW TURBINE AREA actuates
- Secondary IA Pressure on PI-1403 is 55 PSIG and lowering

Which one of the following actions must be taken first by the operating crew?

A. Initiate a MANUAL Reactor Trip

B. Place both FW pumps in manual on the motor speed changers

C. Direct an AO to OPEN the Vacuum Pump suction valves (VA-V-5's)

D. Start available air compressors as necessary to raise pressure >60 PSIG

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) With Secondary Instrument Air pressure less than 80 psig and lowering, the crew must enter OP-TM-AOP-028, LOSS OF INSTRUMENT AIR. When pressure on PI-1403 drops below 60 psig action is taken to some primary cooling valves will remain open and then the reactor is tripped (OP-TM-AOP-028, Rev 9 Page 3)		
A.	Initiate a MANUAL Reactor Trip	CORRECT: See above.
B.	Place both FW pumps in manual on the motor speed changers	INCORRECT: Plausible because the Feedwater pumps are being controlled on the air speed changer. Incorrect because the Feedwater pumps are tripped if feedwater control becomes unreliable.
C.	Direct an AO to OPEN the Vacuum Pump suction valves (VA-V-5's)	INCORRECT: Plausible because the VA-V-5s fail closed on a loss of air. Incorrect because this is not the first action that must be taken.
D.	Start available air compressors as necessary to raise pressure >60 PSIG	INCORRECT: Plausible because this action will be (and should be taken) when pressure is above 60 psig. Incorrect because it is not the action that must be taken now.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	065	AA1.05	
	Importance Rating	3.3		

K/A: Loss of Instrument Air: Ability to operate and / monitor the following as they apply to the Loss of Instrument Air: RPS

Proposed Question:	Question #17	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP028-PCO-4	

Question Source:	Bank #	371798	
	Modified Bank #		
	New		

Question History:	Simulator Exam 7	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the relation between a loss of instrument air and when the reactor must me tripped (actuation of the RPS system).

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

18 **ID: 2103867** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of both Main Feedwater Pumps

Subsequently:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the HSPS level setpoint and where are each OTSG EFW valves setpoints indicated?

A. (1) 0" on the startup range
(2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL

B. (1) 25" on the startup range
(2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL

C. (1) 0" on the startup range
(2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC

D. (1) 25" on the startup range
(2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The Heat Sink Protection System (HSPS) will actuate Emergency Feedwater (EFW) when both Main Feedwater Pumps are tripped. (2) The EF-V-30 A/B/C/D valves will control OTSG level at 25" in the SU range when both Main Feedwater Pumps are lost. (3) To clear a setpoint the operator must take the switches to defeat (for whichever actuation occurred) and place the EF-V-30 in manual (TQ-TM-104-644-C001, Rev 2 Pages 38 and 39).

A.	(1) 0" on the startup range (2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL	INCORRECT: (1) Plausible of the examinee believes the setpoint is reset when all of the HSPS switches are in defeat. Incorrect because the respective EF-V-30s must be placed in hand to clear the seal-in relay. (2) Plausible if examinee does not remember the location of the EF-V-30 controllers.
B.	(1) 25" on the startup range (2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL	INCORRECT: (1) See above. (2) Plausible if examinee does not remember the location of the EF-V-30 controllers.
C.	(1) 0" on the startup range (2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC	INCORRECT: (1) Plausible of the examinee believes the setpoint is reset when all of the HSPS switches are in defeat. Incorrect because the respective EF-V-30s must be placed in hand to clear the seal-in relay. (2) Correct Location
D.	(1) 25" on the startup range (2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC	CORRECT: (1) See above. (2) See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E04	2.1.31
	Importance Rating	4.6	

K/A: Inadequate Heat Transfer - Loss of Secondary Heat Sink: Ability to locate control room switches controls and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question:	Question #18	
Technical Reference(s):	TQ-TM-104-644-C001, Rev 2	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	644-GLO-5	
Question Source:	Bank #	
	Modified Bank #	719815

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New			
Question History:	System Exam 9	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know that the correct level setpoint for the EFW valves for a Lack of Heat Transfer condition. Also, they must know the location of the controllers for the valves.

High Cog: This question is high cog because the examinee must know the EFW setpoint for a loss of both MFW pumps and how to clear the memory for and EFW valve setpoint.

Plant Conditions:

- Loss of BOTH FW pumps
- Reactor and Turbine trip

Event:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the setpoint, from HSPS, for OTSG level control?

A. 0" on the startup range.

B. 25" on the startup range.

C. 50% on the operating range.

D. 75-85% on the operating range.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

19 **ID: 2087129** **Points: 1.00**

Plant Conditions:

- Reactor power was lowered to 50% with ICS in auto

Event:

- ICS T-Ave SETPOINT signal fails ramping HIGHER at 5°F/minute

Which of the following describes the plant response to this malfunction?

A. Reactor power RISES due to rod withdrawal

B. Reactor power is REDUCED due to rod insertion

C. Reactor power DOES NOT CHANGE due to ICS transferring to Tracking Mode

D. Reactor power DOES NOT CHANGE due to the Diamond Transferring to Manual Control

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Tave setpoint failing high (as compared to the Tave actual) a positive error sign will be developed (called Neutron Error) in the Reactor Demand Subsystem of ICS (D553732, Rev P). (2) This positive Neutron Error will cause a rod withdrawal. (3) This malfunction makes it appear that Tave is below the setpoint. The ICS system will raise Tave by pulling the control rods further out, which causes power to rise.

A.	Reactor power RISES due to rod withdrawal	CORRECT: See above.
B.	Reactor power is REDUCED due to rod insertion	INCORRECT: Plausible because this is the effect if the actual Tave signal (rather than the setpoint) began to fail low.
C.	Reactor power DOES NOT CHANGE due to ICS transferring to Tracking Mode	INCORRECT: Plausible if examinee thinks that a large Tave error signal would cause Cross Limits and initiate the Tracking Mode. This would not have an effect on Reactor Power increasing.
D.	Reactor power DOES NOT CHANGE due to the Diamond Transferring to Manual Control	INCORRECT: Plausible if the examinee believes the Diamond would swap to manual control due to a large Neutron Error signal. Incorrect because a large Neutron Error signal does not transfer the Diamond to manual, but it does prevent transferring the Diamond to auto.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	001	AA2.04	
	Importance Rating	4.2		

K/A: Continuous Rod Withdrawal: Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal: Reactor Power and its trend

Proposed Question:	Question #19	
Technical Reference(s):	D553732, Rev P	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	621-GLO-11	

Question Source:	Bank #		
	Modified Bank #	462750	
	New		

Question History:	System Exam 13	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the relation between the ICS Tave setpoint failing high and the effect on the control rods. The control rods will continuously pull to maintain Tave as the setpoint fails high, which raises reactor power.

High Cog: The examinee has to distinguish between the effect of a setpoint failure versus a Tave instrument failure.

Plant Conditions:

- Reactor power is 50% with ICS in automatic.

Sequence of Events:

- ICS T-Ave SETPOINT signal fails slowly, ramping HIGHER at 5°F/minute. due to setpoint potentiometer malfunction.
- ICS control mode does NOT transfer to Tracking during this sequence.

Based on these conditions, identify the ONE statement below that describes initial reactor power trend and control system response to this event.

A. Reactor power RISES due to rod withdrawal.

B. Reactor power is REDUCED due to rod insertion.

C. Reactor power DOES NOT CHANGE due to CRD logic circuit operation.

D. Reactor power DOES NOT CHANGE due to T-Ave control transfer to Feedwater.

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

20 **ID: 2097107** **Points: 1.00**

Plant Conditions:

- Reactor power is 50% with ICS in auto

Event:

- A Group 5 rod drops fully into the core
- MAP G-2-1 CRD PATTERN ASYMMETIC and PPC Alarm L2778 CRD 9" AYSM ROD FAULT come in

Which one of the following describes the manner in which the Group 5 average position indication will include all Group 5 rods?

A. Group 5 average position indication will always include the dropped control rod

B. Group 5 average position indication will automatically reset after the rod is recovered

C. The dropped rod height will be restored to the group average at a 7" delta during the rod recovery

D. The dropped rod height will be restored to the group average at a 9" delta during the rod recovery

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When the absolute position of any CRDM deviates by more than 9" from its group average, and the ASYMMETRIC ROD FAULT BYPASS is false (OP-TM-622-000, Rev 8 Page 48 and 49) a ASYMMETRIC ROD FAULT is issued. Any rod generating an ASYMMETRIC ROD FAULT is removed from the Group Average Calculation. (2) In OP-TM-AOP-062, INOPERABLE ROD (Rev 7 Page 20), the note before Step 1.3 states that at 9" delta the dropped rod height will be restored to the group average calculation causing a small downward step change in group average height.		
A.	Group 5 average position indication will always include the dropped control rod	INCORRECT: Plausible because the examinee could believe that the Group 5 average is simply the average of all the control rods in that group. Incorrect because the group average does not include rods that are greater than 9" asymmetric.
B.	Group 5 average position indication will automatically reset after the rod is recovered	INCORRECT: Plausible because the examinee could know that the group average does not include the dropped rod, but not know the correct method in which the rod gets included into the average. Incorrect because the rod gets included when it is less than 9" apart from the group average.
C.	The dropped rod height will be restored to the group average at a 7" delta during the rod recovery	INCORRECT: Plausible because the examinee could know that the group average does not include the dropped rod, but not know the correct height in which the rod gets included into the average. Incorrect because it gets included in the group average at 9".
D.	The dropped rod height will be restored to the group average at a 9" delta during the rod recovery	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	003	AA2.04
	Importance Rating		

K/A: Dropped Control Rod: Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: Demand position counter and pulse/analog converter

Proposed Question:	Question 20	
Technical Reference(s):	OP-TM-AOP-062, Rev 7	OP-TM-622-000, Rev 8
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	622-GLO-10	
Question Source:	Bank #	
	Modified Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New	X			
Question History:	N/A	Last NRC Exam:	N/A		

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.6	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to the demand position counter and pulse/analog converter (Relative Position Indication at Three Mile Island) operates. They must know that when a rod is being recovered that it will automatically be included back in the group average when it is within 9" of the group average.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

21 **ID: 2095202** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A Control Rod in Group 7 partially drops and is currently 10" below the group average

To prevent lowering power, which one of the following is correct regarding re-aligning the control rod back to its group average in accordance with OP-TM-AOP-062, INOPERABLE ROD?

This control rod must be aligned to the group average within __(1)__.

This time limit restriction is to prevent __(2)__.

A. (1) 1 hour
(2) exceeding tilt limits

B. (1) 1 hour
(2) power peaking and potential fuel damage

C. (1) 2 hours
(2) exceeding tilt limits

D. (1) 2 hours
(2) power peaking and potential fuel damage

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) If a rod were to partially drop into the core (as stated in the stem) the crew would enter OP-TM-AOP-062, INOPERABLE ROD (Rev 7 Page 1) based on the control rod being misaligned greater than 9" from its group. (2) In accordance with OP-TM-AOP-062, Step 3.8 (Rev 7, Page 7) if the rod is not at its in-limit, the rod is misaligned by more than 9 inches and the rod has misaligned by more than 9 inches the crew must reset the relative position indication and align the control rod with its group average. (3) The basis for the one hour time limit in accordance with OP-TM-AOP-062, INOPERABLE ROD BASIS DOCUMENT (Rev 6 Page 7) is to prevent power peaking and potential fuel damage due to lower assembly Xenon levels resulting from power suppression in the assembly with the misaligned rod.

A.	(1) 1 hour (2) exceeding tilt limits	INCORRECT: (1) Correct answer. (2) Plausible because OP-TM-AOP-062 does verify that tilt is in within the limits of the COLR. Incorrect because tilt is not the reason for the 1 hour limit to re-align the control rod.
B.	(1) 1 hour (2) power peaking and potential fuel damage	CORRECT: See above.
C.	(1) 2 hours (2) exceeding tilt limits	INCORRECT: (1) Plausible because the crew must complete a power reduction to less than 60% is the control rod cannot be aligned to its group average. Incorrect because this time limit has to do with the power reduction and not the rod re-alignment. (2) Plausible because OP-TM-AOP-062 does verify that tilt is in within the limits of the COLR. Incorrect because tilt is not the reason for the 1 hour limit to re-align the control rod.
D.	(1) 2 hours (2) power peaking and potential fuel damage	INCORRECT: (1) Plausible because the crew must complete a power reduction to less than 60% is the control rod cannot be aligned to its group average. Incorrect because this time limit has to do with the power reduction and not the rod re-alignment. (2) Correct answer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	005	AK1.06	
	Importance Rating	2.9		

K/A: Inoperable/Stuck Control Rod: Knowledge of the operational implications of the following concepts as they apply to Inoperable/Stuck Control Rod: Bases for power limit for rod misalignment.

Proposed Question:	Question #21	
Technical Reference(s):	OP-TM-AOP-062, Rev 7	OP-TM-AOP-062, Rev 6
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP062-PCO-1	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	b.6		
	55.43			

Comments:

KA Match: This question matches the KA because the examinee must know that if a control rod cannot be re-aligned that power will have to be lowered. The examinee must know the bases for the 1 hour limit at the current power level.

22 **ID: 2089477** **Points: 1.00**

Plant Conditions:

- Plant Startup in progress
- NI-12 and NI-12A are out of service
- 1103-8, APPROACH TO CRITICAL is being performed
- Reactor is NOT critical

Event:

- Loss of VBA

Which one of the following correctly describes whether a plant startup can be performed in this condition?

The plant startup ___(1)___.

A. may continue using NI-11A computer indication

B. may continue after NI-11A is cross connected to NI-11

C. may NOT continue because no source range instrumentation is available

D. may NOT continue because only one source range instrumentation is available

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When Vital Bus A (VBA) is lost, the last remaining source range instrument is lost (OP-AOP-015, Rev 11 Page 21). (2) Since the crew is performing 1103-8, APPROACH TO CRITICAL (and the reactor is NOT critical) with one source range instrument that when VBA is lost the crew has no indication of reactor power. (3) Technical Specification 3.5.1.1 (AMD 189 Page 3-27, AMD 247 Page 3-29, and AMD 189 Page 3-30) state that the crew should restore 1 source range instrument to operable to place the unit in HOT SHUTDOWN within 6 additional hours.		
A.	may continue using NI-11A computer indication	INCORRECT: Plausible if examinee thinks NI-11A is powered from a power source other than VBA. Incorrect because NI-11A is powered from VBA.
B.	may continue after NI-11A is cross connected to NI-11	INCORRECT: Plausible since NI-11A is the installed backup to NI-11 and has to be physically cross connected to perform this function. Incorrect because NI-11A is powered from VBA.
C.	may NOT continue because no source range instrumentation is available	CORRECT: See above.
D.	may NOT continue because only one source range instrumentation is available	INCORRECT: Plausible if the examinee believes there are more source range instruments available. Incorrect because neither source range instrument has power.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	032	AK2.01
	Importance Rating	2.7	

K/A: Loss of Source Range Nuclear Instrumentation: Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies including proper switch positions.

Proposed Question:	Question #22	
Technical Reference(s):	OP-TM-AOP-015, Rev 11	Technical Specification 3.5.1 AMD 247 and 189)

Proposed References to be provided to applicants during examination:	None

Learning Objective:	623-GLO-4	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: The question matches the KA because the examinee must know the power supply to the Source Range Instruments and the operational effects of losing power.

High Cog: This question is high cog because the examinee must know the effect of a loss of VBA will have on a reactor startup with NI-12 out of service.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

23 **ID: 2097109** **Points: 1.00**

During a reactor startup what indication do the operators have to verify the source range and intermediate range instruments are tracking correctly.

Intermediate range indication must come on scale by the time source range indication reads __(1)__ CPS, and track for at least __(2)__ decade(s).

A. (1) 1E5
(2) 1

B. (1) 1E5
(2) 2

C. (1) 1E6
(2) 1

D. (1) 1E6
(2) 2

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The source range indication covers a reactor power from 10E-1 to 10E6 CPS. (2) In accordance with 1103-8, APPROACH TO CRITICALITY (Rev 55 Page 1) states that the overlap between the source range and intermediate range shall be greater than 1 decade. (3) On Step 4.11.7 (Page 6) requires the intermediate range to be at 10E-11 amps by the time the source range is 10E5 cps.		
A.	(1) 1E5 (2) 1	CORRECT: See above.
B.	1) 1E5 (2) 2	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that 2 decades is the requirement. Incorrect because 1 decade is the requirement.
C.	(1) 1E6 (2) 1	INCORRECT: (1) Plausible because this is a reading on the source range instruments. Incorrect because this is the top of the scale and the proper overlap would not be observed if the intermediate range did not come on scale until then. (2) Correct answer.
D.	(1) 1E6 (2) 2	INCORRECT: (1) Plausible because this is a reading on the source range instruments. Incorrect because this is the top of the scale and the proper overlap would not be observed if the intermediate range did not come on scale until then. (2) Plausible if the examinee believes that 2 decades is the requirement. Incorrect because 1 decade is the requirement.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	033		AA2.12
	Importance Rating	2.5		

K/A: Loss of Intermediate Range Nuclear Instrumentation: Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Maximum allowable channel disagreement

Proposed Question:	Question #23	
Technical Reference(s):	1103-8, Rev 55	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	GOP-003-PCO-5	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.6
	55.43	

Comments:

KA Match: This question matches the KA because proper overlap is one of the intermediate range characteristics that operators at Three Miles Island use to verify that an intermediate range instruments are working correctly. If the intermediate range instrument comes on scale too early, that would indicate to an operator that the channel is outside of its max allowable band, and thus not a good instrument to use. If the operators cannot verify overlap on the intermediate range instruments then all of the intermediate range instruments are lost.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

24 **ID: 2107617** **Points: 1.00**

Plant conditions:

- The reactor is at 100% power with ICS in full auto
- The inner door of the Containment Personnel Airlock fails and cannot be closed
- All required Technical Specification actions are complied with

Event:

- An urgent entry into the Containment becomes necessary

Which of the following identifies the action necessary to ensure the Containment Integrity requirements are satisified?

A. The reactor must be shutdown to hot standby prior to opening the outer door.

B. The outer door may be opened provided it is immediately closed after passage.

C. The RCS must be cooled to less than 200°F, with pressure reduced to less than 500 psig prior to opening the outer door.

D. A temporary containment must be set up at the Auxiliary Building entrance from the Reactor Building prior to opening the outer door.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS (Rev 94 Page 7), at least one door in each of the personnel or emergency air locks shall be closed and sealed during personnel passage through these air locks. (2) Since the Personnel Hatch inner door has failed, one door of the personnel or emergency air lock may be open for maintenance, repair or modification provided the other door of the air lock is verified closed within 1 hour, locked within 24 hours, and verified to be locked closed monthly. This action is assumed to have occurred since it is given that all Technical Specification action has been complied with. (2) With the personnel or emergency air lock door interlock mechanism inoperable, entry and exit is permissible under the control of a dedicated individual. Therefore, if an emergency entry is required, the opening of the outer door is to be controlled by a dedicated individual.

A.	The reactor must be shutdown to hot standby prior to opening the outer door.	INCORRECT: Plausible because some of the Containment Technical Specifications, such as building internal pressure, are dependent upon the operating status of the reactor. The operator may incorrectly believe that this Technical Specification is also dependent on the same.
B.	The outer door may be opened provided it is immediately closed after passage.	CORRECT: See above.
C.	The RCS must be cooled to less than 200°F, with pressure reduced to less than 500 psig prior to opening the outer door.	INCORRECT: Plausible because the examinee could believe that if containment integrity is not required then the outer door can be open. Incorrect because the requirement to relax containment is below 300 psig, not 500 psig.
D.	A temporary containment must be set up at the Auxiliary Building entrance from the Reactor Building prior to opening the outer door.	INCORRECT: Plausible because the examinee could believe that a temporary containment could be setup to prevent the spread of any contamination which may be in the Reactor Building. Incorrect because no procedure directs the use of temporary containment outside of the Auxiliary Building.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	069	AA2.02	
	Importance Rating	2.8		

K/A: Loss of Containment: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Verification of automatic and manual means of restoring integrity.

Proposed Question:	Question #24	
Technical Reference(s):	1101-3, Rev 94	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	240-GLO-14	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #	881713		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how to restore containment integrity with a malfunctioning inner RB door.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

25 **ID: 2088596** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power
- EG-Y-1A failed to automatically start and was successfully started manually from the control room
- The following indications are available in the main control room for EG-Y-1A:

Frequency	60.2 Hz
Voltage	4275 Volts
Load	3.3 MWe

In accordance with OP-TM-861-901, DIESEL GENERATOR EG-Y-1A EMERGENCY OPERATIONS, what actions must the CRO take to correct the out of specification reading on EG-Y-1A?

A. Adjust governor to lower EG-Y-1A load

B. Place the exciter in manual and lower voltage from the control room

C. Shutdown non-essential ES bus components to lower EG-Y-1A load

D. Direct the NLO at the diesel to lower the setting on the unit voltage rheostat in the "A" EDG alarm panel

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) That 3.3 MWe is above the rated load for EG-Y-1A. (2) The corrective action in accordance with OP-TM-861-901, DIESEL GENERATOR EG-Y-1A EMERGENCY OPERATIONS (Rev 19, Page 3) is to shutdown non-essential loads to get less than 3.0 MWe.		
A.	Adjust governor to lower EG-Y-1A load	INCORRECT: Plausible because EG-Y-1A is operating above normal frequency. Incorrect because the frequency is within the band of 59-61 hertz per OP-TM-861-901. In addition adjusting the governor will lower frequency and not load.
B.	Place the exciter in manual and lower voltage from the control room	INCORRECT: Plausible because EG-Y-1A is operating above normal voltage. Incorrect since the voltage is within the correct band of 4100V to 4300V per OP-TM-861-901. In addition, this action will not lower the load on the diesel generator.
C.	Shutdown non-essential ES bus components to lower EG-Y-1A load	CORRECT: See above.
D.	Direct the NLO at the diesel to lower the setting on the unit voltage rheostat in the "A" EDG alarm panel	INCORRECT: Plausible because EG-Y-1A is operating above normal voltage. Incorrect since the voltage is within the correct band of 4100V to 4300V per OP-TM-861-901. In addition, this action will not lower the load.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW A05	AA1.3	
	Importance Rating	3.7		

K/A: Emergency Diesel Actuation: Ability to operate an/or monitor the following as they apply to the (Emergency diesel Actuation): Desired operating results during abnormal and emergency situations.

Proposed Question:	Question #25
Technical Reference(s):	OP-TM-861-901, Rev 19
Proposed References to be provided to applicants during examination:	None
Learning Objective:	861-GLO-10

Question Source:	Bank #	805575
	Modified Bank #	
	New	

Question History:	Simulator Exam 7	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know which parameters are out of specification and the appropriate action to take (or what to operate) for the out of specification. The examinee must identify that the diesel generator is operating overloaded and that the crew must shutdown (operate) non-essential loads

High Cog: This question is high cog because the examinee must analyze the parameters in the stem and identify the out of specification. In addition, the examinee must know the procedural action to correct the out of specification.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

26 **ID: 2098967** **Points: 1.00**

In accordance with Technical Specification 3.14.2, FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY, at _______ at Unit1 Screen House, the unit must be placed in HOT STANDBY.

A. 296.0 feet

B. 302.0 feet

C. 350,000 CFS

D. 640,000 CFS

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Technical Specification 3.14.2 FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY (AMD 157 Page 3-60) direct the unit to be brought to hot standby the river reaches 302 feet, which corresponds to 1,000,000 cfs river flow.		
A.	296.0 feet	INCORRECT: Plausible because OP-TM-AOP-002, FLOOD requires the until to be shutdown at this level. Incorrect because this is not the technical specification limit.
B.	302.0 feet	CORRECT: See above.
C.	350,000 CFS	INCORRECT: Plausible because this is an entry criteria into OP-TM-AOP-002. Incorrect because this is not the technical specification limit.
D.	640,000 CFS	INCORRECT: Plausible because there is an action in OP-TM-AOP-002 at 640,000 CFS. Incorrect because this action is to return to normal when it is this CFS and lowering.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW A07	2.2.22	
	Importance Rating	4.0		

K/A: Flooding: Knowledge of limiting conditions for operations and safety limits

Proposed Question:	Question #26	
Technical Reference(s):	OP-TM-AOP-002, Rev 15	TS 3.14.2, Amd 157
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-002-PCO-4	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.4
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the limiting condition for operation flooding.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

27 **ID: 2087792** **Points: 1.00**

In accordance with Rule 2 HPI THROTTLING, high pressure injection must be throttled when __(1)__.

A. subcooling margin is greater than or equal to 250F to prevent violating the reactor vesses pressure-temperature limit

B. subcooling margin is > 25F AND HPI Cooling is NOT required to prevent subcooling margin from becoming excessive

C. subcooling margin is >25F, HPI Cooling is required, and incore temperature is lowering to prevent excessive RCS cooldown

D. when makeup pump flow is greater than 500 gpm to prevent the makeup pump from operating above its design flowrate

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) in Rule 2, HPI THROTTLING of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS (Rev 20 Page 2) Step 2 directs the operators to verify Subcooling Margin (SCM) is less than 250F. (2) If it is not the operators must throttle HPI to control SCM to less than 250F. (3) The basis in accordance with OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT (Rev 11 Page 10), Step A2 is to prevent failure of the reactor vessel by violating the reactor vessel pressure-temperature limit.

A.	subcooling margin is greater than or equal to 250F to prevent violating the reactor vesses pressure-temperature limit	CORRECT: See above.
B.	subcooling margin is > 25F AND HPI Cooling is NOT required to prevent subcooling margin from becoming excessive	INCORRECT: Plausible because Rule 2 has criteria where HPI may be throttled in this condition. Incorrect because HPI does not have to be throttled in this condition. Not until SCM reaches 250F does HPI have to be throttled.
C.	subcooling margin is >25F, HPI Cooling is required, and incore temperature is lowering to prevent excessive RCS cooldown	INCORRECT: Plausible because Rule 2 has criteria where HPI may be throttled in this condition. Incorrect because HPI does not have to be throttled in this condition. Not until SCM reaches 250F does HPI have to be throttled.
D.	when makeup pump flow is greater than 500 gpm to prevent the makeup pump from operating above its design flowrate	INCORRECT: Plausible because this close to the limit of 515 gpm. Incorrect because HPI does not have to be throttled in this condition.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW E13	EK3.02	
	Importance Rating	3.2		

K/A: EOP Rules and Enclosures: Knowledge of the reasons for the following responses as they apply to the (EOP Rules): Normal, abnormal and emergency operating procedures associated with (EOP Rules)

Proposed Question:	Question #27	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	OP-TM-EOP-0101, Rev 11
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	HPI-PCO-1	
Question Source:	Bank #	
	Modified Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New	X			
Question History:	N/A		Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	b.2			
	55.43				

Comments:

KA Match: This question matches the KA because the examinee must know the basis behind the HPI throttling step in Rule 2 (EOP Rule). Rule 2 sends the operator to 900 series procedure (abnormal operating procedure) in order to throttle.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

28 **ID: 2084731** **Points: 1.00**

Which of the following will prevent a Reactor Coolant Pump from starting?

A. Seal Injection below 32 gpm

B. ICCW total flow below 900 gpm

C. 7kV Bus Voltage below 6.8kV

D. Motor Oil Lift System Pressure is below 1000 psig

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know (1) the interlocks which are required to be met to start are Reactor Coolant Pump. OP-TM-226-000, REACTOR COOLANT PUMPS (Rev 14 Page 14) gives a list of the interlocks that must be met. (2) All the distractors at or below their normal value.

A.	Seal Injection below 32 gpm	INCORRECT: Plausible because low seal injection will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 22 gpm.
B.	ICCW total flow below 900 gpm	INCORRECT: Plausible because low ICCW flow will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 550 gpm.
C.	7kV Bus Voltage below 6.8kV	INCORRECT: Plausible because low 7kV bus voltage will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 6.15 kV.
D.	Motor Oil Lift System Pressure is below 1000 psig	CORRECT: Correct answer. Motor Oil Lift System pressure must be above 1000 psig.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	003	K4.03	
	Importance Rating	2.5		

K/A: Reactor Coolant Pump: Knowledge of RCPS design feature(s) and/or interlocks which provide for the following: Adequate lubrication of the RCP

Proposed Question:	Question #28
Technical Reference(s):	OP-TM-226-000, Rev 14
Proposed References to be provided to applicants during examination:	None
Learning Objective:	226-GLO-5

Question Source:	Bank #	502404
	Modified Bank #	
	New	

Question History:	Simulator Exam 5	Last NRC Exam:	N/A

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the motor oil starting interlock for a Reactor Coolant Pump, which is an interlock to ensure adequate lubrication of a Reactor Coolant Pump.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

29 **ID: 2103861** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor trip due to high RCS pressure
- Failed open Pressurizer safety valve resulted in ES actuation
- 1P 480V Bus trip due to electrical fault at the time of the ES actuation

Based on these conditions identify the ONE selection below that completes the following statement:

_________ will OPEN due to the ES actuation signal.

A. BWST suction valve MU-V-14A, ONLY

B. BWST suction valve MU-V-14B, ONLY

C. Both BWST suction valves MU-V-14A and MU-V-14B

D. Both Decay Heat Pump Piggy-back Valves DH-V-7A and DH-V-7B

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The power supply to the BWST suction valve MU-V-14B is 1B 480V ES Valves MCC, which is powered from the 1S 480V ES Bus (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING (Rev 240 Page 53)). (2) The power supply to the BWST suction valve MU-V-14A and the DH pump B to Makeup Pump valve DH-V-7B is 1A ES MCC, which is powered from 1P 480V ES MCC (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING (Rev 240 Page 46)). The 1P 480V ES MCC loses power in the stem, so neither of these valves move position.

A.	BWST suction valve MU-V-14A, ONLY	INCORRECT: Plausible if the examinee reverses the power supply between MU-V-14A and MU-V-14B.
B.	BWST suction valve MU-V-14B, ONLY	CORRECT: See above
C.	Both BWST suction valves MU-V-14A and MU-V-14B	INCORRECT: Plausible if the examinee believes both MU-V-14A and B have power.
D.	Both Decay Heat Pump Piggy-back Valves DH-V-7A and DH-V-7B	INCORRECT: Plausible because DH-V-7A and DH-V-7B provide possible flowpaths from the BWST using the DHP pumps. Incorrect because based on the given conditions, neither valve would open.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	K2.05	
	Importance Rating	2.7		

K/A: Chemical and Volume Control: Knowledge of bus power supplies to the following: MOVs.

Proposed Question:	Question 29	
Technical Reference(s):	1107-4, Rev 240	

Proposed References to be provided to applicants during examination: None

Learning Objective:	642-GLO-8	

Question Source:	Bank #	363623	
	Modified Bank #		
	New		

Question History:	Simulator Exam 5	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.8

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that MU-V-14A will not open due to the loss of the 1P 480V Bus.

High Cog: This question is high cog because the examinee must know the power supply to the each of the MU-V-14's and that on an ES actuation that the valves will open.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

30 **ID: 2084732** **Points: 1.00**

Plant conditions:

- Reactor is at cold shutdown condition
- DHR Train A is operating
- DCCW flow through the DHR cooler is throttled to maintain the RCS at 130°F

Event:

- Total loss of Instrument Air (0 psig)

Which of the following statements describes the response of the Decay Closed Cooling System and the effect on RCS temperature?

A. Closure of DC-V-2A (Cooler inlet) results in RCS heatup

B. Opening of DC-V-2A (Cooler inlet) results in RCS cooldown

C. Closure of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS heatup

D. Opening of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS cooldown

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Due to the loss if instrument air, that the both the inlet/bypass (DC-V-2A/65A) valves to the operating Decay Heat Closed Cooling Cooler would go to their fail safe position (full cooling) (2) The fail safe position for DC-V-2A is full open, and the fail safe position for DC-V-65A is closed. This is referenced in OP-TM-AOP-028, LOSS OF INSTRUMENT AIR (Rev 9 Page 27) Attachment 6.1		
A.	Closure of DC-V-2A (Cooler inlet) results in RCS heatup	INCORRECT: Plausible if the examinee believes that DC-V-2A fails closed. Incorrect because it fails open.
B.	Opening of DC-V-2A (Cooler inlet) results in RCS cooldown	CORRECT: See above.
C.	Closure of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS heatup	INCORRECT: Plausible if the examinee believes that both valves (DC-V-2A/65A) have the same failure method. Incorrect because DC-V-2A fails open and DC-V-65A fails closed.
D.	Opening of DC-V-65A (Cooler bypass) AND DC-V-2A (Cooler inlet) results in RCS cooldown	INCORRECT: Plausible if the examinee believes that both valves (DC-V-2A/65A) have the same failure method. Incorrect because DC-V-2A fails open and DC-V-65A fails closed.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K6.03	
	Importance Rating	2.5		

K/A: Residual Heat Removal: Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR Heat Exchanger

Proposed Question:	Question #30	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	543-GLO-8	

Question Source:	Bank #	355400	
	Modified Bank #		
	New		
Question History:	System Exam 8	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	b.8	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge the effects that a malfunction on the decay closed heat exchanger inlet valve will have on the decay heat system.

High Cog: The examinee has to determine the effect of Loss of IA has on both the inlet valve (DC-V-2A) and the bypass valve (DC-V-65) and the effect of this failure has on the RCS.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The indication provided in the question measures flow at the in the HPI lines prior to the cross connects and MU-V-16s (Drawing 302661, Rev 63), so each set of indicators is measuring actual HPI flow from the Makeup Pumps running on that side. (2) In this lineup, MU-P-1A and B flow is measured by the HPI flows for MU-V-16A and MU-V-16B, and only MU-P-1C flow is measured by the HPI flows for MU-V-16C and MU-V-16D. (3) Rule2, HPI THROTTLING in OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS (Rev 20 Page 4) verifies MU PUMP FLOW is less than 515 gpm/pump, and if it is not then the operator must throttle flow to maintain the flow less than 515 gpm. (4) In this question, the flow through MU-V-16A and B is approximately 540 gpm total (270 gpm/pump) but the flow through MU-V-16C and D is 540 gpm total.						
A.	All MU-V-16s	INCORRECT: Plausible if the examinee determines that flow through the MU-V-16C and D exceeds the limit but believes that all MU-V-16s must be throttled to balance flow in all of the HPI nozzles. Incorrect because the HPI lines are cross connected (A and C, B and D) downstream of the indicators so throttling all MU-V-16s is not required.				
B.	No MU-V-16s	INCORRECT: Plausible if the examinee believes that the makeup pump flow is within the limits of Rule 2. Incorrect because the MU-P-1C is too high.				
C.	MU-V-16A and/or MU-V-16B	INCORRECT: Plausible if the examinee believes that only the A side is above the limit or if MU-P-1B were aligned to the B ES train.				
D.	MU-V-16C and/or MU-V-16D	CORRECT: See above.				
Examination Outline Cross-reference:			Level	RO		SRO
			Tier #	2		
			Group #	1		
			K/A #	006		A1.09
			Importance Rating	2.8		
K/A: Emergency Core Cooling: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pump Amperage, including start, normal and locked.						
Proposed Question:	Question #31					
Technical Reference(s):	OP-TM-EOP-010, Rev 20		Flow Diagram 302-661, Rev 63			
Proposed References to be provided to applicants during examination:				None		
Learning Objective:	EOP-010-PCO-1					
Question Source:	Bank #					
	Modified Bank #					
	New	X				

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments:

This question matches the KA because the examinee must identify that MU-P-1C flow is too high. Flow is proportional to the pump amperage and is the primary indicator in the control room that a makeup pump may be overloaded. After the examinee identifies that MU-P-1C flow is too high, the examinee must know which MU-V-16s that must be throttled to maintain MU-P-1C within limits.

High Cog: This question is high cog because the examinee must determine the Makeup Pump flows from the given indications and determine which is over the limit.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

32 **ID: 2089313** **Points: 1.00**

Plant Conditions:

- Reactor is 100% with ICS in full auto

Event:

- A small Pressurizer steam space leak occurs through a Pressurizer Safety Valve.
- PPC point A0459, RC DRAIN TANK WDL-T-3 TEMP, is in alarm

Which of the following describes the drain pump operation and location where the relief valves discharges to?

The RCDT Pump, WDL-P-8, will __(1)__.
If pressure continues to rise, WDG-V-1 (RCDT relief valve) will open to relieve pressure to the __(2)__.

A. (1) start automatically
(2) Reactor Building

B. (1) start automatically
(2) Waste Disposal Gas Vent Header

C. (1) be manually started
(2) Reactor Building

D. (1) be manually started
(2) Waste Disposal Gas Vent Header

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When a Pressurizer Safety Valve leaks by, steam will be directed to the Reactor Coolant Drain Tank (RCDT), which will heat up the existing water in the tank. (2) The alarm PPC point A0459 comes in when RCDT temperature is greater than 120F. In accordance with the alarm response OP-TM-PPC-A0459, RC DRAIN TANK WDL-T-3 TEMP (Rev 0 Page 1) the RCDT cooler outlet valve, IC-V-20, opens and the RCDT Drain Pump, WDL-P-8 starts automatically when temperature reaches 110F. (3) The examinee must also know that if the pressure in the RCDT gets too high, the relief valve will lift and pressure will be relieved to the Waste Gas Header, which is a system outside of the containment.

A.	(1) start automatically (2) Reactor Building	INCORRECT: Plausible because if the pressure in the RCDT gets too high the rupture disk will burst and relieve pressure to the reactor building. Incorrect because the relief valve lifts to the Waste Gas Header.
B.	(1) start automatically (2) Waste Disposal Gas Vent Header	CORRECT: See above.
C.	(1) be manually started (2) Reactor Building	INCORRECT: Plausible because if the pressure in the RCDT gets too high the rupture disk will burst and relieve pressure to the reactor building. Incorrect because the relief valve lifts to the Waste Gas Header.
D.	(1) be manually started (2) Waste Disposal Gas Vent Header	INCORRECT: Plausible because some pumps and valves must be manually started. Incorrect because WDL-P-8 auto starts.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007	K3.01
	Importance Rating	3.3	

K/A: Pressurizer Relief/Quench Tank: Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

Proposed Question:	Question #32	
Technical Reference(s):	OP-TM-PPC-A0459, Rev 0	302-694, Rev 46
	209-172, Rev 3	

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-2

Question Source:	Bank #	
	Modified Bank #	1149467
	New	

Question History:	N/A	Last NRC Exam:	14-01	(unmodified)

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that where the PRTS will relieves to if the pressure gets too high. The high pressure in the PRTS will open a containment isolation valve to relieve to the vent header which is in the Auxiliary Building.

High Cog: This question is high cog because the examinee must identify the setpoint and determine the plant response.

Plant Conditions:

- The plant is operating at 100% reactor power.
- No evolutions are in progress.

Event:

- A Pressurizer steam space leak occurs through a Pressurizer Safety Valve.
- PPC point A0459, RC Drain Tank WDL-T-3 Temp, is in alarm.

Given the above information, MU-V-17, Normal Makeup to RCS Control Valve, will travel to full ___(1)___ and WDL-P-8, Reactor Coolant Drain Tank Pump, ___(2)___ running.

A. (1) open
(2) is

B. (1) open
(2) is NOT

C. (1) closed
(2) is

D. (1) closed
(2) is NOT

Answer C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

33 **ID: 2104650** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Intermediate Closed Cooling Radiation Monitor, RM-L-9, indication is high at 1E5 CPM due to an earlier RCS leak into ICCW that has been isolated
- RM-L-7, Plant Effluent Rad Monitor, is in service

Event:

- MAP C-3-2, IC SURGE TANK LEVEL HI/LO ACTUATED

Currently:

- Intermediate Cooling Surge Tank, IC-T-1, level is **7** inches, LOWERING at 1 inch per minute
- RM-L-7, Plant Effluent Rad Monitor, reading is slowly rising

Which one of the following describes the location of the leak and the procedure that must be entered for this condition?

A. (1) Inside one of the Letdown Coolers, MU-C-1A/B
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

B. (1) Inside on of the Intermediate Service Coolers, IC-C-1A/B
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

C. (1) Inside one of the Letdown Coolers, MU-C-1A/B
(2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE

D. (1) Inside on of the Intermediate Service Coolers, IC-C-1A/B
(2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The plant conditions that an RCS leak into the ICCW has already occurred (and been isolated) and the RM-L-9 reading is still high. (2) When MAP-C-3-2 comes in and RM-L-7 is rising, the examinee should determine that there is a leak from ICCW in the Intermediate Service Cooler IC-C-1A/B. (3) In addition, the entry criteria for OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING (Rev 5 Page 1) have been met.

A.	(1) Inside one of the Letdown Coolers, MU-C-1A/B (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	INCORRECT: (1) Plausible because these coolers are cooled by Intermediate Closed Cooling Water. Incorrect because counts are going up in the plant discharge monitor, so the leak must be in the Intermediate Service Cooler. (2) Correct Answer.
B.	(1) Inside on of the Intermediate Service Coolers, IC-C-1A/B (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	CORRECT: See above.
C.	(1) Inside one of the Letdown Coolers, MU-C-1A/B (2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE	INCORRECT: (1) Plausible because these coolers are cooled by Intermediate Closed Cooling Water. Incorrect because counts are going up in the plant discharge monitor, so the leak must be in the Intermediate Service Cooler. (2) Plausible because OP-TM-AOP-050 covers RCS leaks in various components but would not be the correct procedures to enter in this time.
D.	(1) Inside on of the Intermediate Service Coolers, IC-C-1A/B (2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE	INCORRECT: (1) Correct answer. (2) Plausible because OP-TM-AOP-050 covers RCS leaks in various components but would not be the correct procedures to enter in this time.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	008	A2.04	
	Importance Rating	3.3		

K/A: Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PRMS alarm

Proposed Question:	Question #33	
Technical Reference(s):	OP-TM-AOP-033, Rev 5	OP-TM-MAP-C0302, Rev 4
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-032-PCO-1	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #		
	Modified Bank #	363656	
	New		

Question History:	unmod on Simulator Exam 6	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee will have to analyze the impact that the rising reading on the liquid radiation monitor has on the plant. The examinee will must identify the lead location from the ICCW surge tank level lowering and RM-L-7 reading rising. The examinee will have to choose the correct AOP to implement to mitigate the consequences.

High Cog: Examinee has to determine the proper location of the leak base on the given plant conditions and then determine the correct procedure flowpath.

Plant Conditions:

- Reactor is operating at 100% power with ICS in full automatic.
- Intermediate Cooling Pump, IC-P-1A, is operating.
- Both Intermediate Service Coolers, IC-C-1A and IC-C-1B, are in service.
- Both Letdown Coolers, MU-C-1A and MU-C-1B, are in service.
- Nuclear Services River Water Pumps, NR-P-1A and NR-P-1C, are operating.
- Intermediate Closed Cooling Radiation Monitor, RM-L-9, indication at 1E5 CPM due to an earlier RCS leak into ICCW that has been isolated.
- RM-L-7, Plant Effluent Rad Monitor, is in service.

Event:

- MAP C-3-2, IC Surge Tank Level Hi/Lo actuated.
- Currently:
  - Intermediate Cooling Surge Tank, IC-T-1, level is 7 inches, LOWERING at 1 inch per minute.
  - RM-L-7 reading is slowly rising.

Based on these conditions, identify the ONE selection below that describes the location of the leak.

A. Inside one of the Letdown Coolers.

B. CRD cooling outlet pipe inside the RB.

C. Intermediate Service Cooler, IC-C-1A/B.

D. Inside one of the Reactor Coolant Pump Thermal Barrier Heat Exchangers.

Answer C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

34 **ID: 2089093** **Points: 1.00**

Plant Conditions:

- Reactor Trip from 100% power
- OP-TM-EOP-001, REACTOR TRIP IMA's and VSSV's are complete
- Pressurizer level is 98" and the setpoint 100"

Event:

- Selected Pressurizer Level instrument starts to slowly fail low

Assuming no operator actions, MU-V-17, RCS Makeup Valve, will __(1)__ and __(2)__.

A. (1) open
(2) pressurizer heaters will deenergize due to low level interlock

B. (1) open
(2) additional pressurizer heaters will energize as RCS pressure lowers

C. (1) close
(2) pressurizer heaters will deenergize due to low level interlock

D. (1) close
(2) additional pressurizer heaters will energize as RCS pressure lowers

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When the pressurizer level instrument starts to fail low, that MU-V-17 will open to try to maintain level at the setpoint and open up. (TQ-TM-104-211-C001 (Rev 11 Page 43)). (2) There is an interlock which deenergizes the pressurizer heaters when the pressurizer level indicates less than 80". (TQ-TM-104-220-C001 (Rev 9 Page 38)).

A.	(1) open (2) pressurizer heaters will deenergize due to low level interlock	CORRECT: See above.
B.	(1) open (2) additional pressurizer heaters will energize as RCS pressure lowers	INCORRECT: (1) Correct (2) Plausible because pressurizer level will lower. When pressurizer level lowers, pressure lowers as well. Incorrect because while pressurizer level will indicate lower, as MU-V-17 will open and pressurizer level will actually go up.
C.	(1) close (2) pressurizer heaters will deenergize due to low level interlock	INCORRECT: (1) Plausible because it the setpoint signal were failing low the MU-V-17 would close. Incorrect because actual level is failing low, not the setpoint signal. (2) Correct Answer.
D.	(1) close (2) additional pressurizer heaters will energize as RCS pressure lowers	INCORRECT: (1) Plausible because it the setpoint signal were failing low the MU-V-17 would close. Incorrect because actual level is failing low, not the setpoint signal (2) Plausible because pressurizer level will lower. When pressurizer level lowers, pressure lowers as well. Incorrect because while pressurizer level will indicate lower, as MU-V-17 will open and pressurizer level will actually go up.

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

K/A: Pressurizer Pressure Control: Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PZR LCS

Proposed Question:	Question #34	
Technical Reference(s):	OP-TM-211-472, Rev 4	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	211-GLO-11	

Question Source:	Bank #		
	Modified Bank #	354888	
	New		

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question History:	System Exam 11	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the relationship between pressurizer level, the setpoint, and the level that the pressurizer heaters cutout.

High Cog: This question is high cog because the examinee must analyze the malfunction and determine the plant response and identify an interlock that actuates.

Plant Conditions:

- 100% Power.

Event:

- The controlling pressurizer level channel fails high.

Assuming no operator action has occurred, which of the following statements describes the plant response?

The Pressurizer Level Control Valve, (MU-V-17), ___(1)___ and ___(2)___.

A. (1) opens
(2) actual level rises and spray valve opens

B. (1) closes
(2) actual pressurizer level and RCS pressure lower

C. (1) position remains the same
(2) actual pressurizer level and RCS pressure rise

D. (1) position remains the same
(2) actual pressurizer level and RCS pressure lower

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

35 **ID: 2104682** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- NI-7 is failed high

Event:

- MAP A-1-6 INVERTER FAILED alarm comes in
- RPS Channel C cabinet lights are all dark

Which one of the following statements describes the required actions?

A. Ensure reactor power is reduced to less than 75%

B. Ensure the plant is stabilized and enter OP-TM-AOP-017, LOSS OF VBC

C. Ensure the reactor trips and perform the IMA's of OP-TM-EOP-001, REACTOR TRIP

D. Ensure the plant is stabilized and enter OP-TM-AOP-027 Loss of ATA or ICS Auto Power

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) With the combination of MAP A-1-6 and RPS Channel C lights becoming dark that Vital Bus C was lost. (2) NI-7, which is already failed high, will also lose power because its power supply is NI-7. The plant will still have 3 operable NI's and also meet the required degree of redundancy.

A.	Ensure reactor power is reduced to less than 75%	INCORRECT: Plausible because when a vital bus is lost the RPS will only see three Reactor Coolant Pumps in operation. The examinee could believe that a runback occurs and/or the correct action is to lower power less than 75%. Incorrect because those actions are not required.
B.	Ensure the plant is stabilized and enter OP-TM-AOP-017, LOSS OF VBC	CORRECT: See above.
C.	Ensure the reactor trips and perform the IMA's of OP-TM-EOP-001, REACTOR TRIP	INCORRECT: Plausible because the initial plant conditions have NI-7 failed high, which would give one RPS channel trip. Losing any other Vital Bus would have resulted in an automatic reactor trip. Incorrect because NI-7 is powered from Vital Bus C, so when VBC was lost no further RPS channel trip was generated.
D.	Ensure the plant is stabilized and enter OP-TM-AOP-027 Loss of ATA or ICS Auto Power	INCORRECT: Plausible if the examinee believes that VBC is the power supply to ATA and ICS auto. Incorrect because it is not.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	A2.07	
	Importance Rating	3.6		

K/A: Reactor Protection: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power

Proposed Question:	Question #35	
Technical Reference(s):	OP-TM-AOP-017, Rev 10	
	OP-TM-MAP-A0106, Rev 0	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-017-PCO-2	

Question Source:	Bank #	1738330	
	Modified Bank #		
	New		
Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how the loss of Vital Bus C will affect the plant and the procedure used to mitigate the malfunction. The loss of the vital bus will cause the RPS channel to de-energize, the examinee must understand the impact and the procedure entry which is required.

High Cog: Examinee has to analyze the given plant indications and then determine the correct course of action based on these conditions.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

36 **ID: 2088359** **Points: 1.00**

In the event of a reactor coolant system rupture, operation of _______ will limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad.

A. 2 Decay Heat Removal Pumps, 1 Core Flood Tank

B. 2 Decay Heat Removal Pumps, 2 Core Flood Tanks

C. 1 Makeup Pump, 1 Decay Heat Removal Pump, 1 Core Flood Tank

D. 1 Makeup Pump, 1 Decay Heat Removal Pump, 2 Core Flood Tanks

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) With one train of ES available (Makeup Pump and a Decay Heat Removal Pump) and both Core Flood tanks will protect the core. (2) Each of these components has restrictive technical specifications which will require a plant shutdown. (3) The Technical Specification 3.3 basis defines the required components to limit peak cladding temperature to less than 2,200F and the metal-water reaction to that representing less than 1 percent of the clad.

A.	2 Decay Heat Removal Pumps, 1 Core Flood Tanks	INCORRECT: Plausible since one of the ECCS criteria is to ensure long term cooling, having both Decay Heat Removal Pumps would help ensure this criteria is met. Incorrect because a Makeup Pump and both Core Flood Tanks are required.
B.	2 Decay Heat Removal Pumps, 2 Core Flood Tanks	INCORRECT: Plausible since one of the ECCS criteria is to ensure long term cooling, having both Decay Heat Removal Pumps would help ensure this criteria is met. Incorrect because a Makeup Pump and both Core Flood Tanks are required..
C.	1 Makeup Pump, 1 Decay Heat Removal Pump, 1 Core Flood Tanks	INCORRECT: Plausible because the examinee could believe that Core Flood is a redundant system (i.e. only one tank is required). Incorrect because both Core Flood Tanks are required.
D.	1 Makeup Pump, 1 Decay Heat Removal Pump, 2 Core Flood Tanks	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	K3.01
	Importance Rating	4.4	

K/A: Engineered Safety Features Actuation: Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Proposed Question:	Question #36	
Technical Reference(s):	Technical Specification 3.3, AMD 290	

Proposed References to be provided to applicants during examination:	None
Learning Objective:	642-GLO-8

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge that even with a loss of major components of the ECCS system, the minimum required components to show fuel protection are still available.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

37 **ID: 2084746** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Reactor Building Cooling Fan, AH-E-1C, is shutdown and the control switch is in NORMAL AFTER STOP

Event:

- Manual reactor trip due to a Small Break LOCA inside Containment
- ESAS High RB Pressure Bistable operation:

Time	Action
T = 0	Channel 1 Bistable trips
T = + 5 seconds	Channel 2 Bistable trips
T = + 10 seconds	Channel 3 Bistable trips

Which one of the following identifies the time in which AH-E-1C will start?

AH-E-1C will start __(1)__ seconds after the __(2)__ ES channel actuates.

A. (1) 5
(2) first

B. (1) 5
(2) second

C. (1) 10
(2) second

D. (1) 15
(2) first

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) ES components need to see two ES signals to actuate. (2) When ES actuates, block loading initiates. Block 1 loads start immediately, then Block 2 loads start 5 seconds later. The RB Cooling Fans trip on Block 1, then restart in slow on Block 2. (TQ-TM-104-740-C001, Rev 7 Page 23).		
A.	(1) 5 (2) first	INCORRECT: Plausible if the examinee believes that ES components initiate the Block Loading sequence after the 1st channel actuates. The AH-E-1's would start 5 seconds after the actuation. Incorrect because Block Loading starts after 2 channels actuate.
B.	(1) 5 (2) second	CORRECT: See above.
C.	(1) 10 (2) second	INCORRECT: Plausible if the examinee believes that Block 2 will trip the AH-E-1s first then start them later. In addition, the examinee could believe the AH-E-1's start after block 3. Incorrect because they start 5 seconds after block two.
D.	(1) 15 (2) first	INCORRECT: Plausible because this time would coincide with 5 seconds after Block 3. Incorrect the AH-E-1s start 5 seconds after Block 2.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	022	A3.01	
	Importance Rating	4.1		

K/A: Containment Cooling: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

Proposed Question:	Question #37	
Technical Reference(s):	TQ-TM-104-824-C001, Rev 8	TQ-TM-104-740-C001, Rev 7
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	824-GLO-10	

Question Source:	Bank #	586874	
	Modified Bank #		
	New		

Question History:	Simulator Exam 1	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have the knowledge to know when the containment ventilation fans start after an ES actuation. Each of the bistables will illuminate lights on the ES status panel, when two bistables are actuated, the containment cooling fan will start after a time delay.

High Cog: This question is high cog because the examinees must analyze the ES actuation and determine when the RB Emergency Cooling fans start.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

38 **ID: 2087162** **Points: 1.00**

Which of the following has the ability to provide backup power automatically to Building Spray Pump, BS-P-1A?

EG-Y-4 = Station Blackout Diesel Generator

A. 4 Bus

B. EG-Y-1A

C. EG-Y-1B

D. EG-Y-4

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The normal power supply to BS-P-1A is 1D 4160V ES Bus (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING REV 221 Page 18). (2) The automatic backup to the 1D 4160V bus is EG-Y-1A.

A.	4 Bus	INCORRECT: Plausible if the examinee believes that 1D 4160V bus is powered from the 8 bus. Incorrect because the normal power supply is the 4 bus.
B.	EG-Y-1A	CORRECT: See above.
C.	EG-Y-1B	INCORRECT: Plausible if the examinee believes the back up to the 1D 4160V Bus is EG-Y-1B. Incorrect because it is not.
D.	EG-Y-4	INCORRECT: Plausible if the examinee believes that EG-Y-4 can automatically provide backup power to the 1D 4160V bus. Incorrect because EG-Y-4 can only be manually aligned to power the 1D 4160V bus.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026	K2.01
	Importance Rating	3.4	

K/A: Containment Spray: Knowledge of bus power supplies to the following: Containment spray pumps

Proposed Question:	Question #38	
Technical Reference(s):	1107-4, Rev 221	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	214-GLO-4	
Question Source:	Bank #	
	Modified Bank #	719780
	New	
Question History:	N/A	Last NRC Exam: N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the bus power supplies to the containment spray pumps.

719780

Which ONE of the following power supplies has the ability to provide backup power AUTOMATICALLY to the 1B Reactor Building Spray Pump (BS-P-1B)?

A. 4 Bus

B. 1A Emergency Diesel Generator (EG-Y-1A)

C. 1B Emergency Diesel Generator (EG-Y-1B)

D. Station Blackout Diesel Generator (EG-Y-4)

Answer C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

39 **ID: 2104742** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor Trip

What OTSG pressure setpoint will the Turbine Bypass Valves be maintaining and what is the basis of that pressure setpoint?

Turbine Bypass Valves open to control Turbine Header Pressure at ___(1)___ and the basis for this setpoint is to ___(2)___.

A. (1) 960 psig
(2) prevent exceeding Main Steam System design pressure

B. (1) 960 psig
(2) limit the Pressurizer level reduction from the cooldown of reactor coolant

C. (1) 1010 psig
(2) prevent exceeding Main Steam System design pressure

D. (1) 1010 psig
(2) limit the Pressurizer level reduction from the cooldown of reactor coolant

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) On a reactor trip, the ICS selects a +125 bias to adjust the turbine header pressure (TQ-TM-104-411-C001 MAIN STEAM, Rev 8 Page 30). (2) The basis of raising pressure is to limit the RCS cooldown and shrink.

A.	(1) 960 psig (2) prevent exceeding Main Steam System design pressure	INCORRECT: (1) Plausible because the examinee could believe that the OTSG pressure stays the same for a reactor trip. Incorrect because the turbine bypass valves control at a higher pressure. (2) Plausible because by maintaining pressure at this level the Main Steam System design pressure will not be exceeded. Incorrect because ensuring Main Steam System design pressure is not exceeded is the function of the Main Steam Safety Valves.
B.	(1) 960 psig (2) limit the Pressurizer level reduction from the cooldown of reactor coolant	INCORRECT: (1) Plausible because the examinee could believe that the OTSG pressure stays the same for a reactor trip. Incorrect because the turbine bypass valves control at a higher pressure. (1) See above.
C.	(1) 1010 psig (2) prevent exceeding Main Steam System design pressure	INCORRECT: (1) Correct Setpoint (2) Plausible because by maintaining pressure at this level the Main Steam System design pressure will not be exceeded. Incorrect because ensuring Main Steam System design pressure is not exceeded is the function of the Main Steam Safety Valves
D.	(1) 1010 psig (2) limit the Pressurizer level reduction from the cooldown of reactor coolant	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A1.05	
	Importance Rating	3.2		

K/A: Main and Reheat Steam: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: RCS T-ave

Proposed Question:	Question #39	
Technical Reference(s):	TQ-TM-104-411-C001, Rev 8	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	411-GLO-5	
Question Source:	Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	Modified Bank #	1142251		
	New			
Question History:	N/A	Last NRC Exam:	14-01	Unmodified

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the Main Steam Turbine Bypass setpoint for a reactor trip and the basis. The basis is to minimize the pressurizer level shrink, which can be directly tied to RCS T-ave. The turbine bypass valves maintain OTSG pressure at 1010 psig, which will maintain RCS T-ave at approximately 555F, which will limit the outsurge of the Pressurizer. The design limit this prevents exceeding is losing Pressurizer Level indication on a reactor trip.

High Cog: This question is high cog because the examinee must analyze what will happen to the turbine bypass valve setpoint on a reactor trip.

1142251
Plant Conditions:

- Plant is at 100% power.

Event:

- A Reactor Trip occurs due to a loss of a Reactor Coolant Pump without a proper ICS runback.

Given the above information, the Turbine Bypass Valves open to control Turbine Header Pressure at ___(1)___ and the Atmospheric Dump Valves will ___(2)___.

A. (1) 960 psig
(2) open fully at 1040 psig

B. (1) 960 psig
(2) begin to open at 1026 psig and be fully open at 1052 psig

C. (1) 1010 psig
(2) open fully at 1040 psig

D. (1) 1010 psig
(2) begin to open at 1026 psig and be fully open at 1052 psig

Answer	D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

40 **ID: 2084753** **Points: 1.00**

REFEENCE PROVIDED

Plant conditions:

- Reactor power is 90% with ICS in auto

Event:

- A high pressure feedwater heater has just been removed from service

Which of the following statements describes the IMMEDIATE ICS response to these conditions when FW temperature lowers?

A. The Feedwater Demand signal would be modified to RAISE feedwater flow

B. The Feedwater Demand signal would be modified to LOWER feedwater flow

C. Steam Generator BTU limits would be reached AND the unit would be placed in TRACK

D. Feedwater to Reactor Crosslimits would be reached AND the unit would be placed in TRACK

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When the heater string is removed from service, the actual temperature of the feedwater (as seen by the ICS Feedwater Subsystem) will be lower. (2) At this point, to maintain the heat balance the Feedwater Demand Signal will be modified lower as less is needed because the temperature is colder (to maintain the proper BTU exchange between the primary and secondary). (TQ-TM-104-621-C001, Rev 10 Page 39 and 40)		
A.	The Feedwater Demand signal would be modified to RAISE feedwater flow	INCORRECT: Plausible if the examinee believes that more feedwater will be required because the plant is less efficient. Incorrect because less feedwater is needed.
B.	The Feedwater Demand signal would be modified to LOWER feedwater flow	CORRECT: See above
C.	Steam Generator BTU limits would be reached AND the unit would be placed in TRACK	INCORRECT: Plausible because FW Temperature is an input to the BTU Limit Circuit. Incorrect because the FW Temperature circuit would lower actual FW demand to maintain FW temperature above BTU Limits. Also, BTU limits does not place ICS in Track.
D.	Feedwater to Reactor Crosslimits would be reached AND the unit would be placed in TRACK	INCORRECT: Plausible because the lower FW temperature would effect the Total FW demand. Total FW demand is compared to actual FW Flow to cause a Feedwater to Reactor Crosslimit. Incorrect because FW demand has to be greater than actual FW flow to cause a Crosslimit. Cross Limits do cause the initiation of Tracking.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059	K1.07
	Importance Rating	3.2	

K/A: Main Feedwater: Knowledge of the physical connections and/or cause effect relationships between the MFW and the following systems: ICS

Proposed Question:	Question #40	
Technical Reference(s):	TQ-TM-104-621-C001, Rev 10	
	D553731, Rev Q	

Proposed References to be provided to applicants during examination:	D553731, Rev Q

Learning Objective:	621-GLO-5	

Question Source:	Bank #	357042	
	Modified Bank #		
	New		

Question History:	System Exam 13	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how ICS controls Main Feedwater.

High Cog: This question is high cog because the examinee must analyze and understand how removing a high pressure feedwater string effects the plant and ICS.

41 **ID: 2104765** **Points: 1.00**

Plant Conditions:

- Reactor Power is 100% with ICS in full auto
- Main Feedwater line leak identified inside "A" D-Ring
- Containment Building pressure is 1.5 psig and stable

Event:

- Containment Building pressure begins to rise
- Manual reactor trip due to RB equipment concerns
- Current plant conditions:
  - OTSG 1A Startup Range level is 4 inches, stable
  - OTSG 1B Startup Range level is 95 inches, lowering at 1-inch per minute
  - Containment Building pressure is stable at 4.2 psig
  - RC-P-1A and RC-P-1B have been tripped

Which of the following identifies HSPS actuation response with regards to (1) Emergency Feedwater pumps, and (2) Emergency Feedwater valves?

A. (1) EF-P-1 & EF-P-2A start, ONLY.
(2) EF-V-30A & EF-V-30C level control setpoint is 25 inches in the STARTUP Range; EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.

B. (1) EF-P-1 & EF-P-2A start, ONLY.
(2) EF-V-30A & EF-V-30C level control setpoint is 50% in the OPERATE Range. EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.

C. (1) EF-P-1, EF-P-2A, & EF-P-2B start.
(2) EF-V-30A/B/C/D all control at 25 inches in the STARTUP Range.

D. (1) EF-P-1, EF-P-2A, & EF-P-2B start.
(2) EF-V-30A/B/C/D level control setpoint is 50% in the OPERATE Range.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) One of the functions of the Heat Sink Protection System (HSPS) is to start Emergency Feedwater (EFW) on the following signals: Loss of both Main Feedwater Pumps, Loss of all Reactor Coolant Pumps, Reactor Building Pressure > 4 psig, and less than 10" in the Startup Range in either OTSG. The OTSG level setpoint 25" for all actuations with the exception of loss of all the Reactor Coolant Pumps which is 50% in the Operating Range. (MAP J Rev 49 Page 2). (2) HSPS will actuate on 4 psig in the Reactor Building and and Low 'A' OTSG level and start all the EFW pumps and control level at 25" in the Startup Range.		
A.	(1) EF-P-1 & EF-P-2A start, ONLY. (2) EF-V-30A & EF-V-30C level control setpoint is 25 inches in the STARTUP Range, EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.	INCORRECT: (1) Plausible if the examinee determines that the 'A' OTSG is the only OTSG which needs feedwater. Incorrect because even though the 'A' OTSGs level is low, all of the EFW pumps start. (2) Plausible because EF-V-30A and 30C are controlled by the 'A' train of HSPS. If the examinee believed only 'A' train would actuate its components then this would be correct. Incorrect because that is not how HSPS works.
B.	(1) EF-P-1 & EF-P-2A start, ONLY. (2) EF-V-30A & EF-V-30C level control setpoint is 50% in the OPERATE Range. EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.	INCORRECT: (1) Plausible if the examinee determines that the 'A' OTSG is the only OTSG which needs feedwater. Incorrect because even though the 'A' OTSGs level is low, all of the EFW pumps start (2) Plausible because the Reactor Coolant Pumps in the 'A' loop are tripped. The examinee could believe that the setpoint for the 'A' train components is 50% in the Operating Range. Incorrect because that is not how HSPS works.
C.	(1) EF-P-1, EF-P-2A, & EF-P-2B start. (2) EF-V-30A/B/C/D all control at 25 inches in the STARTUP Range.	CORRECT: See above.
D.	(1) EF-P-1, EF-P-2A, & EF-P-2B start. (2) EF-V-30A/B/C/D level control setpoint is 50% in the OPERATE Range.	INCORRECT: (1) Correct answer. (2) Plausible because the examinee could believe that because the Reactor Coolant Pumps are secured in one loop that the setpoint is 50% in the Operating Range.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K4.02	
	Importance Rating	4.5		

K/A: Auxiliary/Emergency Feedwater: Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: AFW automatic start upon loss of MFW pump, S/G level, blackout, or safety injection

Proposed Question:	Question #41

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Technical Reference(s):	MAP J, Rev 49	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	424-GLO-5	

Question Source:	Bank #	371302	
	Modified Bank #		
	New		

Question History:	Sim Exam 5	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the automatic starting interlocks for the EFW pumps.

High Cog: This question is high cog because the examinee has to analyze conditions in the stem and determine which EFW pumps start and the level setpoint of the EFW control valves.

42 **ID: 2084797** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- 1B Auxiliary Transformer senses high pressure on 2 rate of rise detectors on the Load Tap Changer (LTC)

Which one of the following is the correct response of the plant electrical system?

A. All 7KV and 4KV switchgear transfer to 1A Auxiliary Transformer

B. 86 relays will lockout busses associated with 1B Auxiliary Transformer

C. 7KV and 4KV BOP switchgear transfer to 1A Auxiliary Transformer

D. LTC is locked out, no further changes in electrical plant lineup occur

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The Auxiliary Transformer will trip if the Load Tap Changers (LTC) pressure relays see both of the pressure relays exceed their limit (2 out of 2 logic) (TQ-TM-104-701-C001, Rev 9 Page 20). (2) When the 1B Auxiliary Transformer trips all of the BOP switchgear fast transfers to the 1A Auxiliary Transformer (MAP AA, Rev 41 Page 13).		
A.	All 7KV and 4KV switchgear transfer to 1A Auxiliary Transformer	INCORRECT: Plausible if the examinee believes that all equipment will transfer to the 1A Auxiliary Transformer. Incorrect because the ES equipment gets powered by its Emergency Diesel Generator.
B.	86 relays will lockout busses associated with 1B Auxiliary Transformer	INCORRECT: Plausible since 86 lockout relays will affect breakers associated with the 4 Bus. Incorrect because they will not effect the 1B Auxiliary Transformer breakers.
C.	7KV and 4KV BOP switchgear transfer to 1A Auxiliary Transformer	CORRECT: See above.
D.	LTC is locked out, no further changes in electrical plant lineup occur	INCORRECT: Plausible if examinee does not know that the pressure switches associated with load tap changer will initiate the auto transfer. Incorrect because they will trip the Auxiliary Transformer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	K4.03	
	Importance Rating	2.8		

K/A: AC Electrical Distribution: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers

Proposed Question:	Question #42	
Technical Reference(s):	MAP AA, Rev 41	TQ-TM-104-701-C001, Rev 9
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	731-GLO-11	

Question Source:	Bank #	1006758	
	Modified Bank #		
	New		

Question History:	System Exam Final	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge			

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the actions which occur on an automatic bus transfer. The examinee must know which bus breakers open and then close to maintain power.

High Cog: This question is high cog because the examinee must determine that the malfunction will have an effect on the unit electrical distribution system. In addition, the examinee must know the lineup of the electrical busses after the malfunction.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

43 **ID: 2084804** **Points: 1.00**

Which selection below describes an occurrence that would indicate the existence of a problem in the "B" 125/250VDC Distribution System?

A. MAP K-3-4, MN TURB DC OIL PMP STRT/TRBL actuates

B. CM-V-1 and CM-V-3 fail closed, rendering RM-A-2 inoperable

C. Loss of control power indication for all Reactor Coolant Pumps

D. VA-V-5A and VA-V-5C fail closed (VA-P-1A and C suction valves)

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Under the current condition where the "B" 123/250VDC Distribution System is lost, the crew would enter OP-TM-AOP-024, "B" DC SYSTEM FAILURE. Attachment 7.1 (Rev 7 Page 35) describes the effects of a loss of "B" DC, one of which being loss of control power to all RCP breakers. This would be apparent in the control room by the breaker indications lights above each Reactor Coolant Pump being off.		
A.	MAP K-3-4, MN TURB DC OIL PMP STRT/TRBL, actuates	INCORRECT: Plausible because this alarm comes in from a loss of DC. Incorrect because the loss of "A" DC brings this alarm in.
B.	CM-V-1 and CM-V-3 fail closed, rendering RM-A-2 inoperable	INCORRECT: Plausible because these valves in from are DC powered. Incorrect because the loss of "A" DC brings fails these valves closed.
C.	Loss of control power indication for all Reactor Coolant Pumps	CORRECT: See above
D.	VA-V-5A and VA-V-5C fail closed (VA-P-1A and C suction valves)	INCORRECT: Plausible because these valves in from are DC powered. Incorrect because the loss of "A" DC brings fails these valves closed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	A3.01
	Importance Rating	2.7	

K/A: DC Electrical Distribution: Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

Proposed Question:	Question #43	
Technical Reference(s):	OP-TM-AOP-0241, Rev 8	OP-TM-AOP-024, Rev 7

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-024-PCO-4	

Question Source:	Bank #	1738593	
	Modified Bank #		
	New		

Question History:	Comp 2	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how the indicating lights on the Reactor Coolant Pumps could be used to monitor that the 'B' DC system has had a malfunction. In addition, that automatic operation (i.e. a pump trip) cannot be performed at console center.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

44 **ID: 2094874** **Points: 1.00**

Which Fuel Oil tank(s) meet the Technical Specification requirements for the Emergency Diesel Generators, EG-Y-1A and EG-Y-1B?

A. 550 gallon EG-Y-1A/B Day Tanks, DF-T-2A/B

B. 30k gallon Diesel Generator Fuel Oil Storage Tank, DF-T-1

C. 50k gallon Fuel Oil Storage Tank, FO-T-1

D. 200k gallon Fuel Oil Storage Tank, FO-T-2

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The technical specification diesel fuel oil is at least 25,000 gallons available in storage tank (Tech Spec 3.7.1.e, AMD 224 Page 3-42). (2) In accordance with 1107-3, DIESEL GENERATOR (Rev 153 Page 15) DF-T-1 is required to have 25,000 gallons of usable fuel oil to support emergency diesel generator operations.		
A.	550 gallon EG-Y-1A/B Day Tanks, DF-T-2A/B	INCORRECT: Plausible because each diesel generator has its own day tank. Incorrect because it does not meet the 25,000 gallon requirement.
B.	30k gallon Diesel Generator Fuel Oil Storage Tank, DF-T-1	CORRECT: See above
C.	50k gallon Fuel Oil Storage Tank, FO-T-1	INCORRECT: Plausible because the tank capacity exceeds the 25,000 gallon requirement. Incorrect because this tank does not directly supply the diesel generators.
D.	200k gallon Fuel Oil Storage Tank, FO-T-2	INCORRECT: Plausible because the tank capacity exceeds the 25,000 gallon requirement. Incorrect because this tank does not directly supply the diesel generators.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	064		
	Importance Rating	3.2		

K/A: Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks

Proposed Question:	Question #44	
Technical Reference(s):	1107-3, Rev 153	Technical Specification 3.7.1e AMD 224

Proposed References to be provided to applicants during examination:	None

Learning Objective:	861-GLO-14	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge that a malfunction or unavailability of the 30k gallon Diesel Generator Fuel Oil tank will make the diesel generators inoperable.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

45 **ID: 2084805** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto
- Miscellaneous Waste Evaporator is in service

Event:

- RM-A-5 and RM-A-15, Condenser Vacuum Pump Exhaust Radiation Monitors, suddenly show rising trends
- MAP C-1-1, RADIATION LEVEL HIGH comes in due to RM-A-5 and RM-A-15

Which of the following actions, if performed within 15 minutes of the event, would initially differentiate between a fuel failure and OTSG tube leak in accordance with OP-TM-MAP-C0101?

A. Survey the letdown line to determine if a significant change in RCS activity has occurred.

B. Request Chemistry to obtain a condensate return sample of the evaporator and analyze for boron.

C. Check Letdown Radiation Monitor, RM-L-1, readings to determine if there is a sudden upward trend.

D. Check Letdown Radiation Monitor, RM-L-1, LO readings to determine if there is a sudden upward trend.

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) A note in OP-TM-MAP-C0101 (Rev 4, Page 8) notifies the crew that a change in RCS activity many not be observed on RM-L-1 (Letdown Line Radiation Monitor) until 30 to 60 minutes later due to transport time. (2) The procedure directs the crew to request Rad Con to perform a survey of the letdown line to determine if a significant change in RCS activity. (2) Due to the minute amount of OTSG leakage that may exist, an RCS leak may show up in RM-A-5/15 before RM-L-1.

A.	Survey the letdown line to determine if a significant change in RCS activity has occurred.	CORRECT: See above.
B.	Request Chemistry to obtain a condensate return sample of the evaporator and analyze for boron.	INCORRECT: Plausible because this is a step in OP-TM-MAP-C0101. Incorrect because Rad Con would be sampling for activity, not boron.
C.	Check Letdown Radiation Monitor, RM-L-1, readings to determine if there is a sudden upward trend.	INCORRECT: Plausible because this is a Letdown Monitor. Incorrect because this monitor would take 30 to 60 minutes for a change in reading to be observed.
D.	Check Letdown Radiation Monitor, RM-L-1, LO readings to determine if there is a sudden upward trend.	INCORRECT: Plausible because this is a Letdown Monitor. Incorrect because this monitor would take 30 to 60 minutes for a change in reading to be observed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073	2.1.7
	Importance Rating	4.4	

K/A: Process Radiation Monitoring: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question:	Question #45	
Technical Reference(s):	OP-TM-MAP-C0101, Rev 4	

Proposed References to be provided to applicants during examination:	None
Learning Objective:	661-GLO-12

Question Source:	Bank #	858179	
	Modified Bank #		
	New		

Question History:	Simulator Exam 9	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.11
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the operational implications that an elevated reading on the letdown line would have if RM-A-5/15 were to alarm. This information would differentiate between taking actions to shutdown via MAP-C0101 or OP-TM-EOP-005, OTSG TUBE LEAKAGE.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

46 **ID: 2106163** **Points: 1.00**

REFERENCE PROVIDED

In accordance with OP-TM-541-234, IST OF NR-P-1A AND NSRW VALVES DURING SINGLE PUMP OPERATIONS which areas of Attachment 7.2 correspond to the following limitations?

Area 1 is to (1) .
Area 2 is to (2) .
Area 3 is to (3) .

A. 1) avoid pump wear or damage
2) avoid NR strainer clogging
3) prevent pump runout

B. 1) avoid NR strainer clogging
2) avoid pump wear or damage
3) prevent pump runout

C. 1) avoid pump wear or damage
2) prevent pump runout
3) avoid NR strainer clogging

D. 1) prevent pump runout
2) avoid pump wear or damage
3) avoid NR strainer clogging

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The limits and precautions of OP-TM-541-234 (Rev 7 Page 2) require Nuclear River Water Pumps be run in the operating band of OP-TM-541-000, PRIMARY COMPONENT COOLING (Rev 24A Page 37) Attachment 7.2. The examinee must know the limitations apply to the following areas in the provided reference:
Area 1 - To prevent excessive pump wear or damage, do not operate NR pumps for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.
Area 2 - To avoid clogging of the NR strainers (i.e., keep strainer pressure >20 psig), do not operate for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.
Area 3 - To prevent NR pump run-out if a NR pump trips when two NR pumps were operating, then maintain NSRW pressure above “two pump operation limit” on Attachment 7.2 on PI-217 (CC).

A.	1) avoid pump wear or damage 2) avoid NR strainer clogging 3) prevent pump runout	CORRECT: See above.
B.	1) avoid NR strainer clogging 2) avoid pump wear or damage 3) prevent pump runout	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.
C.	1) avoid pump wear or damage 2) prevent pump runout 3) avoid NR strainer clogging	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.
D.	1) prevent pump runout 2) avoid pump wear or damage 3) avoid NR strainer clogging	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076	2.2.12
	Importance Rating	3.7	

K/A: Service Water System: Knowledge of surveillance procedures

Proposed Question:	Question #46	
Technical Reference(s):	OP-TM-541-000, Rev 24	OP-TM-541-234, Rev 7
Proposed References to be provided to applicants during examination:		Edited Attachment 7.2 of OP-TM-541-000

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	531-GLO-10				
Question Source:	Bank #				
	Modified Bank #				
	New	X			
Question History:	N/A		Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	b.7			
	55.43				

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of a surveillance procedure of a service water system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

47 **ID: 2104876** **Points: 1.00**

Sequence of Events:

- A rupture occurs in the 1 1/2 to 3/4 inch reducer on the air header supplying MS-V-3D.
- IA-P-4 trips on overload.
- IA-P-1A/B and SA-P-1A/B are started successfully.
- Instrument air primary (PI-222) and secondary (PI-1403) pressure indicators on PL are tracking together and continue to lower.

Which of the following indications would be observed on PL when IA-V-26 (Secondary Plant IA Supply Valve) closes?

When header pressure drops below:

A. 60 psig then PI-222 starts to RISE and PI-1403 continues to LOWER

B. 60 psig then PI-1403 starts to RISE and PI-222 continues to LOWER

C. 80 psig then PI-222 starts to RISE and PI-1403 continues to LOWER

D. 80 psig then PI-1403 starts to RISE and PI-222 continues to LOWER

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) MS-V-3D is in the Turbine Building, which will be isolated when IA-V-26 closes at 60 psig header pressure. (2) When IA-V-26 is closed, that will isolate the secondary instrument air from the primary instrument air. PI-1402 will continue to lower and PI-222 will being to rise.		
A.	60 psig then PI-222 starts to RISE and PI-1403 continues to LOWER.	CORRECT: See above.
B.	60 psig then PI-1403 starts to RISE and PI-222 continues to LOWER	INCORRECT: Plausible if location of air leak is misunderstood. Incorrect instrument header responses would be opposite.
C.	80 psig then PI-222 starts to RISE and PI-1403 continues to LOWER	INCORRECT: Plausible because 80 psig is a common instrument air setpoint. Incorrect because it is not the correct setpoint for IA-V-26.
D.	80 psig then PI-1403 starts to RISE and PI-222 continues to LOWER.	INCORRECT: Plausible because 80 psig is a common instrument air setpoint. Incorrect because it is not the correct setpoint for IA-V-26.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	078	A4.01	
	Importance Rating	3.1		

K/A: Instrument Air: Ability to manually operate and/or monitor in the control room: Pressure gauges

Proposed Question:	Question #47	
Technical Reference(s):	302271, Rev 73	302268, Rev 18

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-028-PCO-5	

Question Source:	Bank #	353834	
	Modified Bank #		
	New		

Question History:	Simulator Exam 8	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how the pressure gauges in the control room respond during a loss of instrument air.

High Cog: This question is high cog because the examinee must know the setpoint for IA-V-26 and analyze the plant response for the location of the leak.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

48 **ID: 2104885** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Design Basis LOCA occurs
- RB Pressure is currently 35 psig and rising
- All "A" Train of ESAS fails to automatically actuate
- No operator actions have occurred

In accordance with ___(1)___, the crew must manually initiate "A" train ___(2)___ in order to close all containment isolation valves.

A. (1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) 30# ESAS, only

B. (1) OP-TM-EOP-006, LOCA COOLDOWN
(2) 4# and 30# ESAS

C. (1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) 4# and 30# ESAS

D. (1) OP-TM-EOP-006, LOCA COOLDOWN
(2) 30# ESAS, only

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When equipment operates properly, there are separate containment isolation valves that close on a 4# ES and 30# ES signal. (2) In addition, the operator must know that the 30# ES signal is not a redundant signal to the 4# ES, specifically that the 30# ES is the only signal that closes NS-V-4/15/35, IC-V-2/3/4/6, MU-V-25/26 The 30# ES will close none of the valves that the 4# ES closes and vice versa. (3) The procedure that directs the actuation of the 4# and 30# ES signals is OP-TM-EOP-006, LOCA COOLDOWN (Rev 15 Page 3) steps 3.1 and 3.5.		
A.	(1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) 30# ESAS, only	INCORRECT: (1) Plausible because OP-TM-AOP-050 covers RCS leakage and does have a step for ES actuation. Incorrect because operators initiate 1600# ES, not the 4# or 30#. (2) Plausible if the examinee believes that the 30# ES is redundant to the 4# ES in that the 30# will close the same valves. Incorrect because the 30# ES does not close any of the valves that the 4# ES does.
B.	(1) OP-TM-EOP-006, LOCA COOLDOWN (2) 4# and 30# ESAS	CORRECT: See above
C.	(1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) 4# and 30# ESAS	INCORRECT: (1) Plausible because OP-TM-AOP-050 covers RCS leakage and does have a step for ES actuation. Incorrect because operators initiate 1600# ES, not the 4# or 30#. (2) Correct answer.
D.	(1) OP-TM-EOP-006, LOCA COOLDOWN (2) 30# ESAS, only	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that the 30# ES is redundant to the 4# ES in that the 30# will close the same valves. Incorrect because the 30# ES does not close any of the valves that the 4# ES does

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103	A2.03	
	Importance Rating	3.5		

K/A: Containment: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions: Phase A and B isolation

Proposed Question:	Question #48	
Technical Reference(s):	OP-TM-EOP-006, Rev 13	MAP E, Rev 61
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EOP006-PCO-4	
Question Source:	Bank #	
	Modified Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New	X			
Question History:	N/A		Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know that the crew must initiate BOTH 4# and 30# 'A' ES to fully isolate the containment and start the building spray pumps. Many containment isolation valves close on 4# ES but some Nuclear Service, Intermediate Closed Cooling Water, and Makeup valves close on a 30# ES. The terminology 'Phase A and B' most close resemble these 4# and 30# ES signals.

High Cog: This question is high cog because the examinee must know that to completely isolate the Reactor Building that both the 4# and 30# signals are needed. In addition, the examinee must determine that the crew would be in the OP-TM-EOP-006 procedure for a LOCA cooldown based on plant conditions.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

49 **ID: 2084744** **Points: 1.00**

Plant Conditions:

- Reactor Coolant (RCS) pressure 2166 psig.
- Reactor Coolant Drain Tank (RCDT) pressure is 10 psig .
- The PORV is stuck partially open.

Given the above information, what is the approximate temperature and phase of the fluid downstream of the PORV?

A. 193°F and Saturated.

B. 240°F and Saturated.

C. 498°F and Superheated.

D. 648°F and Superheated.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When fluid leaks past the PORV that it maintains a constant enthalpy. (2) Due to this process and the pressure drop from the Pressurizer to the Reactor Coolant Drain Tank (RCDT) the temperature will lower to the saturation temperature of the pressure in the RCDT.		
A.	193°F and Saturated	INCORRECT: Plausible if the candidate incorrectly interprets the Mollier.
B.	240°F and Saturated.	CORRECT: See above.
C.	498°F and Superheated.	INCORRECT: Plausible if the candidate incorrectly interprets the Mollier.
D.	648°F and Superheated.	INCORRECT: Plausible if the candidate incorrectly interprets the Mollier.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	010	K5.02	
	Importance Rating	2.6		

K/A: Pressurizer Pressure Control: Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Constant enthalpy expansion through a valve

Proposed Question:	Question #49	
Technical Reference(s):	Steam Tables / Mollier Diagram	

Proposed References to be provided to applicants during examination:	Steam Tables

Learning Objective:	220-PCO-5	

Question Source:	Bank #	909285	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments:
KA Match: This question matches the KA because the examinee must know that leakage past the PORV is a constant enthalpy expansion process.
High Cog: This question is high cog because the examinee must use the mollier diagram to get the correct temperature and quality of steam.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

50 **ID: 2087160** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- 1E 125/250V DC ES Distribution Panel is de-energized to support ground-busting

Which one of the following describes the impact on CRD breaker operation if an automatic RPS reactor trip signal is generated during these conditions?

A. Breakers CRD-CB-A and CRD-CB-C will NOT open
Breakers CRD-CB-B and CRD-CB-D will open

B. Breakers CRD-CB-A and CRD-CB-D will NOT open
Breakers CRD-CB-B and CRD-CB-C will open

C. Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-D
All CRD breakers will open

D. Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-C
All CRD breakers will open

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When an RPS generates two trip signals, all of the CRD breakers (CRD-CB-A/B/C/D) should open. (2) To ensure they open the breakers are designed with a shunt trip coil and an UV relay. (3) The shunt trips must be powered from 125 VDC to open the breaker. The shunt signal is generated from RPS. (4) CRD-CB-A and CRD-CB-C are powered from 1E 125 VDC. CRD-CB-B and CRD-CB-D are powered from 1F 125VDC (TQ-TM-014-622-C001, Rev 7 Page 29 and 30). (4) All breakers will open from the UV relays.

A.	Breakers CRD-CB-A and CRD-CB-C will NOT open; Breakers CRD-CB-B and CRD-CB-D will open	INCORRECT: Plausible because the shunt trip from CRD-CB-A and CRD-CB-B are not operable. The examinee could believe they will not open. Incorrect because the UV coils will open the breakers.
B.	Breakers CRD-CB-A and CRD-CB-D will NOT open; Breakers CRD-CB-B and CRD-CB-C will open	INCORRECT: Plausible if the examinee does not know which DC supplies power to each CRD breaker. Incorrect because all CRD breakers open.
C.	Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-D; All CRD breakers will open	INCORRECT: Plausible if the examinee does not know which DC supplies power to each CRD breaker. Incorrect because the CRD-CB-A and CRD-CB-C shunt trips will not operate.
D.	Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-C; All CRD breakers will open	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	K1.02
	Importance Rating	3.4	

K/A: Reactor Protection: Knowledge of the physical connections/or cause effect relationships between the RPS and the following systems: 125V dc system

Proposed Question:	Question #50	
Technical Reference(s):	OP-TM-AOP-024, Rev 7	OP-TM-AOP-0241, Rev 8
	TQ-TM-104-622-C001, Rev 7	

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-4

Question Source:	Bank #	371279
	Modified Bank #	
	New	

Question History:	N/A	Last NRC Exam:	N/A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the physical connection between the RPS and 125V DC system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

51 **ID: 2087140** **Points: 1.00**

Event:

- A normal plant cooldown in progress in accordance with 1102-11, PLANT COOLDOWN
- MAP D-3-8, CF-V-1A/B POSITION ABNORMAL alarm is received
- CF-V-1A/1B breakers are closed
- RCS pressure is 645 psig and lowering

What (1) Control Room indications are available to verify this alarm and, (2) which action(s) must be taken IAW MAP D-3-8?

A. (1) ES Status Panel indication, ONLY
(2) Close CF-V-1A and CF-V-1B prior to reaching 600 psig

B. (1) Control Console indication, ONLY
(2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B

C. (1) ES Status Panel and Control Console indication
(2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B

D. (1) ES Status Panel and Control Console indication
(2) Close CF-V-1A and CF-V-1B prior to reaching 600 psig

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1102-11, PLANT COOLDOWN (Rev 156 Page 22) the CF-V-1 (Core Flood Tank Outlet Valves) may be closed when the reactor is shutdown and RCS pressure is between 2200 - 750 psig. (2) In this case the MAP alarm D0308 will come in when RCS pressure is less than 650 psig and CF-V-1A and/or CF-V-1B are open. (3) OP-TM-MAP-D0308 (Rev 3, Page 1) directs RCS pressure to be stabilized or close CF-V-1A and CF-V-1B. (4) CF-F-1A and CF-V-1B have indicating lights on the ES status panel and Console Center.		
A.	(1) ES Status Panel indication, ONLY (2) Close CF-V-1A and CF-V-1B prior to reaching 600 psig	INCORRECT: (1) Plausible because the indicating lights for CF-V-1A/B on Console Center are off due to the breakers being open at power. Incorrect because at this point in the cooldown the breakers would have been closed and indicating light power available. (2) Plausible because this is the nominal pressure for the Core Flood tanks to empty into the RCS. Incorrect procedural guidance directs stabilizing the plant or closing the valves.
B.	(1) Control Console indication, ONLY (2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B	INCORRECT: (1) Plausible because the indicating lights for CF-V-1A/B on Console Center are off due to the breakers being open at power. Incorrect because at this point in the cooldown the breakers would have been closed and indicating light power available. (2) Correct answer.
C.	(1) ES Status Panel and Control Console indication (2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B	CORRECT: See above.
D.	(1) ES Status Panel and Control Console indication (2) Close CF-V-1A and CF-V-1B prior to reaching 600 psig	INCORRECT: (1) Correct answer. (2) Plausible because this is the nominal pressure for the Core Flood tanks to empty into the RCS. Incorrect procedural guidance directs stabilizing the plant or closing the valves.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006	A4.02
	Importance Rating	4.0	

K/A: Emergency Core Cooling: Ability to manually operate and/or monitor in the control room: Valves

Proposed Question:	Question #51	
Technical Reference(s):	OP-TM-MAP-D0308, Rev 3	TQ-TM-104-213-C001, Rev 5
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	213-GLO-10	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #	371767		
	Modified Bank #			
	New			
Question History:	Simulator Exam 6	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the indication and manual operation of valves in an ES system (Core Flood) during a plant cooldown.

High Cog: This question is high cog because the examinee must know the reason the alarm D-3-8 came in and analyze the plant conditions to determine correct position of the valves.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

52 **ID: 2107217** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A Small Break LOCA causes a Reactor Trip on Low Pressure
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER is entered due to a Main Feedwater upset that occurred after the Reactor Trip.
- EFW is manually initiated in accordance with OP-TM-424-901, EMERGNECY FEEDWATER

(1) After the Foxboro controllers are back in automatic, what actions must the operators take to have automatic control of the EF-V-30's at the 25" setpoint?
(2) What is the plant cooldown rate limit for this casualty?

The EF-V-30 setpoint lever would be placed in __(1)__ and set the thumbwheel to 25.
The cooldown rate limit is __(2)__.

A. (1) L (local)
(2) 100F/hr

B. (1) R (Remote)
(2) 100F/hr

C. (1) L (Local)
(2) 50F/hr

D. (1) R (Remote)
(2) 50F/hr

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Enclosure 4 of 1105-19, HEAT SINK PROTECTION SYSTEM (Rev 30 Page 45) describes the use of the EF-V-30 controllers. The R/L switch selects either remote (auto setpoint) or local (manual setpoint). The thumbwheel adjusts the local setpoint. (2) The question does not explicitly imply that any automatic EFW actuation setpoint was exceeded so using the controller is the only way to put the EF-V-30's in automatic. (3) In accordance with OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS, Guide 11, COOLDOWN RATE LIMITS (Rev 20 Page 24) the cooldown rate is 100F/hr.

A.	(1) L (local) (2) 100F/hr	CORRECT: (1) See above
B.	(1) R (Remote) (2) 100F/hr	INCORRECT: (1) Plausible because that is the other option on the switch. Incorrect because the Remote setpoint would be at 0 since no EFW actuation setpoint was exceeded. (2) Correct answer.
C.	(1) L (Local) (2) 50F/hr	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that OP-TM-EOP-004 directs securing all Reactor Coolant Pumps. Incorrect because it only directs securing one in each loop.
D.	(1) R (Remote) (2) 50F/hr	INCORRECT: (1) Plausible because that is the other option on the switch. Incorrect because the Remote setpoint would be at 0 since no EFW actuation setpoint was exceeded. (2) Plausible if the examinee believes that OP-TM-EOP-004 directs securing all Reactor Coolant Pumps. Incorrect because it only directs securing one in each loop.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061		A3.02
	Importance Rating	4.0		

K/A: Auxiliary/Emergency Feedwater: Ability to monitor automatic operation of the AFW, including: RCS cooldown during AFW operations.

Proposed Question:	Question #52	
Technical Reference(s):	1105-19, Rev 30	OP-TM-EOP-010, Rev 20
Proposed References to be provided to applicants during examination:		None
Learning Objective:	644-GLO-5	
Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how to place EFW control in automatic control and the cooldown limits based on plant conditions.

High Cog: This question is high cog because the examinee must know how the EFW controller works and how to correctly set the lever. Also, the examinee must analyze plant conditions to determine the cooldown rate limit.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

53 **ID: 2084785** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A fire in a cable chase has caused a short in CO-V-9A (FW-P-1A SUCTION VALVE) causing it to go CLOSED

What is the Feedwater System response to this event?

__(1)__ will trip due to __(2)__

A. (1) FW-P-1A
(2) CO-V-9A closing

B. (1) FW-P-1A
(2) low suction pressure

C. (1) Both Main Feedwater Pumps
(2) CO-V-9A closing

D. (1) Both Main Feedwater Pumps
(2) low suction pressure

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The closure of CO-V-9A is an automatic trip for FW-P-1A (OP-TM-MAP-M0101, FWP 1A TRIP, Rev 2, Page 1)

A.	(1) FW-P-1A (2) CO-V-9A closing	CORRECT: See above.
B.	(1) FW-P-1A (2) low suction pressure	INCORRECT: Plausible if examinee recognizes CO-V-9 closing would cause low suction pressure. Incorrect because there is no low suction pressure trip for a Main Feedwater Pump (MFP)
C.	(1) both Main Feedwater Pumps (2) CO-V-9A closing	INCORRECT: Plausible because there is a common suction header for both MFPs and closing one CO-V-9 would effect both MFPs. Incorrect because CO-V-9A/B are after the common suction header.
D.	(1) both Main Feedwater Pumps (2) low suction pressure	INCORRECT: Plausible because there is a common suction header for both MFPs and closing one CO-V-9 would effect both MFPs. Incorrect because CO-V-9A/B are after the common suction header.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	059	A3.03	
	Importance Rating	2.5		

K/A: Main Feedwater: Ability to monitor automatic operation of MFW, including: Feedwater pump suction flow pressure

Proposed Question:	Question #53	
Technical Reference(s):	OP-TM-MAP-M0101, Rev 2	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	401-GLO-11	

Question Source:	Bank #	371788	
	Modified Bank #		
	New		

Question History:	System Exam 6	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that a Main Feedwater Pump will trip if the suction valve closes. No indication exists on the control room panel solely dedicated to feedwater pump suction pressure, Condensate Booster Pump discharge pressure and the indication that the suction valve, CO-V-9A/B is open show that the Main Feedwater pumps have proper suction pressure. The examinee must know that a low feedwater suction pressure does not trip the Main Feedwater Pumps, but a closed suction valve will.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

54 **ID: 2087972** **Points: 1.00**

Plant Conditions:

- Reactor power is 50% with ICS in auto

Event:

- RC-P-1A trips

(1) Which oil pump(s) automatically start?
(2) What is the location of the oil pumps indication(s) and control switch(es)?

RC-P-2A = HP Lift Pump
RC-P-3A = Backstop Oil Pump

A. (1) RC-P-2A
(2) Console Right

B. (1) RC-P-2A
(2) Console Center

C. (1) RC-P-2A and RC-P-3A
(2) Console Right

D. (1) RC-P-2A and RC-P-3A
(2) Console Center

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-MAP-F0101, RCP MOTOR TRIP (Rev 0 Page 1) when a Reactor Coolant Pump trips then the operators ensure one RC-P-2 is in service and start at least one RC-P-3. (2) All of the controls are on console center.		
A.	(1) RC-P-2A (2) Console Right	INCORRECT: (1) Correct answer. (2) Plausible because the controls are on the right side of console center, close to Console Right.
B.	(1) RC-P-2A (2) Console Center	CORRECT: See above
C.	(1) RC-P-2A and RC-P-3A (2) Console Right	INCORRECT: (1) Plausible because both pumps must be started in accordance with the alarm response. Incorrect because the RC-P-2 is the only one that auto starts. (2) Plausible because the controls are on the right side of console center, close to Console Right.
D.	(1) RC-P-2A and RC-P-3A (2) Console Center	INCORRECT: (1) Plausible because both pumps must be started in accordance with the alarm response. Incorrect because the RC-P-2 is the only one that auto starts. 2. Correct answer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	2.1.31
	Importance Rating	4.6	

K/A: Reactor Coolant Pump: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

Proposed Question:	Question #54	
Technical Reference(s):	OP-TM-MAP-F0101, Rev 0	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	226-GLO-10	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the location of the control room switches and the correct the oil pumps running for a Reactor Coolant Pump trip.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

55 **ID: 2105725** **Points: 1.00**

Plant conditions:

- The plant is at 14% power
- The Main Generator is being synchronized to the grid, IAW OP-TM-301-102, MAIN TURBINE GENERATOR STANDBY TO OPERATING MODE

In accordance with OP-TM-301-102, generator kilovolts must be __(1)__ than system kilovolts, and the Synchroscope must be going in the __(2)__ direction.

A. (1) greater
(2) slow

B. (1) greater
(2) fast

C. (1) less
(2) slow

D. (1) less
(2) fast

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-301-102, MAIN TURBINE GENERATOR STANDBY TO OPERATING MODE (Rev 26 Page 32) in Step 4.19.9 the operators raise generator kilovolts 4KV higher than the system kilovolt. (2) In step 4.19.10 the operators adjust Turbine speed until the synchroscope is rotating in the fast direction at no more than 1 revolution every 3 seconds.		
A.	(1) greater (2) slow	INCORRECT: (1) Correct answer. (2) Plausible of the examinee believes the synchroscope should be going in the slow (opposite) direction. Incorrect because the synchroscope should be going in the fast direction.
B.	(1) greater (2) fast	CORRECT: See above.
C.	(1) less (2) slow	INCORRECT: (1) Plausible if the examinee believes that the generator voltage should be system voltage. Incorrect because it must be greater to prevent motoring the generator. (2) Plausible of the examinee believes the synchroscope should be going in the slow (opposite) direction. Incorrect because the synchroscope should be going in the fast direction.
D.	(1) less (2) fast	INCORRECT: (1) Plausible if the examinee believes that the generator voltage should be system voltage. Incorrect because it must be greater to prevent motoring the generator. (2) Correct answer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	A4.03	
	Importance Rating	2.8		

K/A: AC Electrical Distribution: Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages.

Proposed Question:	Question #55		
Technical Reference(s):	OP-TM-301-102, Rev 26		
Proposed References to be provided to applicants during examination:	None		
Learning Objective:	711-GLO-6		
Question Source:	Bank #		
	Modified Bank #		
	New	X	
Question History:	N/A	Last NRC Exam:	N/A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have the ability to monitor the correct parameters in the control room to synchronize the main generator to the grid.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

56 **ID: 2105895** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto

Event:

- MAP alarm A-1-6, INVERTER FAILED, illuminates
- MAP alarm A-3-8, INVERTER 1B/1D/1F TROUBLE, illuminates
- 4 psig RB Pressure Channel RB3B indicates actuated
- 500 psig ESAS Channel RC6B indicates actuated
- 1600 psig ESAS Channel RC3B indicates actuated
- 30 psig RB Pressure ESAS Channel RB6B indicates actuated
- Reactor Protection Channel D has no lights illuminated

The plant has experienced a loss of Vital Bus __(1)__, and to find controlling group rod position indication the operators can use __(2)__.

A. (1) C
(2) PPC only

B. (1) C
(2) PPC and the left-side PIP panel

C. (1) D
(2) PPC only

D. (1) D
(2) PPC and the left-side PIP panel

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must recognize: (1) With the given indications that Vital Bus 'D' was lost and that OP-TM-AOP-018, LOSS OF VBD (Rev 9) was entered. Indications of loss of Vital Bus 'D' are in Attachment 1 on Page 13. (2) Step 3.8 allows the operators to select either groups 1-4 or 5-7 on CRD-FPM-A (left side PI Panel). (3) Rod position indication can still be found on the PPC (TQ-TM-104-622-C001, Rev 7 Page 34) when Vital Bus 'D' was lost.		
A.	(1) C (2) PPC only	INCORRECT: (1) Plausible if the examinee identifies the loss of power as Vital Bus 'C'. Incorrect because the power loss is Vital Bus 'D'. Vital Bus 'C' powers the other PIP panel. (2) Plausible if the examinee believes that the only remaining indication for the controlling group rods is the PPC. Initially the left PI panel will show only group 1-4 rods. The left PI panel will not show the controlling group of rods (groups 5-7) until the operators reposition the switch by the PI panel. Incorrect because when the operators reposition the switch the controlling group of rods will be shown on the PI panel.
B.	(1) C (2) PPC and the left-side PIP panel	INCORRECT: (1) Plausible if the examinee identifies the loss of power as Vital Bus 'C'. Incorrect because the power loss is Vital Bus 'D'. Vital Bus 'C' powers the other PIP panel. (2) Correct answer.
C.	(1) D (2) PPC only	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that the only remaining indication for the controlling group rods is the PPC. Initially the left PI panel will show only group 1-4 rods. The left PI panel will not show the controlling group of rods (groups 5-7) until the operators reposition the switch by the PI panel. Incorrect because when the operators reposition the switch the controlling group of rods will be shown on the PI panel.
D.	(1) D (2) PPC and the left-side PIP panel	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014	K3.02
	Importance Rating	2.5	

K/A: Rod Position Indication: Knowledge of the effect that a loss or malfunction of the RPIS will have on the following: Plant Computer

Proposed Question:	Question #56	
Technical Reference(s):	OP-TM-AOP-018, Rev 9	TQ-TM-104-622-C001, Rev 7
	OP-TM-AOP-0181, Rev 5	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-018-PCO-1	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the effect that a malfunction of the P2 processor (which calculates rod position for the PIP) has on the PPC. The examinee must know rod position can still be found on the plant computer.

High Cog: This question is high cog because the examinee must analyze the stem and determine the cause of the indications. In addition, the examinee must determine where Rod Position Indication is available.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

57 **ID: 2084852** **Points: 1.00**

Sequence of Events:

- The reactor has tripped from 100% power and all control rods have inserted.
- Nine minutes after the trip, the following plant conditions exist:
  - Intermediate Range, NI-3, indicates a power level decrease of one decade every three minutes.
  - Intermediate Range, NI-4, indicates a power level decrease of one decade every nine minutes.
  - NI-3 currently reads 8×10^{-10} amps decreasing.
  - NI-4 currently reads 6×10^{-8} amps decreasing.

Which one of the following explains the reason for the response of the intermediate range nuclear instruments?

Compensating voltage on __(1)__ is set too __(2)__.

A. (1) NI-3
(2) low

B. (1) NI-3
(2) high

C. (1) NI-4
(2) low

D. (1) NI-4
(2) high

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must recognize: (1) On a normal reactor trip that the intermediate range power levels should decay off at about 1/3 decade per minute (DPM) (TQ-TM-104-623-C001, Rev 5 Page 17). (2) NI-3 is only instrument that is displaying the correct behavior so the examinee must determine that NI-4 is malfunctioning. (3) In accordance with the TQ-TM-104-623-C001 (Page 17), if NI-4 were overcompensating then the neutron flux would result in a less than actual indication. This is not the case since with indication is higher than expected. NI-4 is undercompensated, which means its compensating voltaget is set too low.		
A.	(1) NI-3 (2) low	INCORRECT: (1) Plausible if the examinee determines that NI-4 is exhibiting the correct behavior after a reactor trip. Incorrect because NI-4 power indication is too high. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.
B.	(1) NI-3 (2) high	INCORRECT: (1) Plausible if the examinee determines that NI-4 is exhibiting the correct behavior after a reactor trip. Incorrect because NI-4 power indication is too high. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.
C.	(1) NI-4 (2) low	CORRECT: See above.
D.	(1) NI-4 (2) high	INCORRECT: (1) Correct answer. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015	K6.01
	Importance Rating	2.9	

K/A: Nuclear Instrumentation: Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Sensors, detectors, and indicators

Proposed Question:	Question #57	
Technical Reference(s):	TQ-TM-104-623-C001, Rev 5	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	623-GLO-11	

Question Source:	Bank #	371840	
	Modified Bank #		
	New		

Question History:	Sim Exam 1	Last NRC Exam:	N/A	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

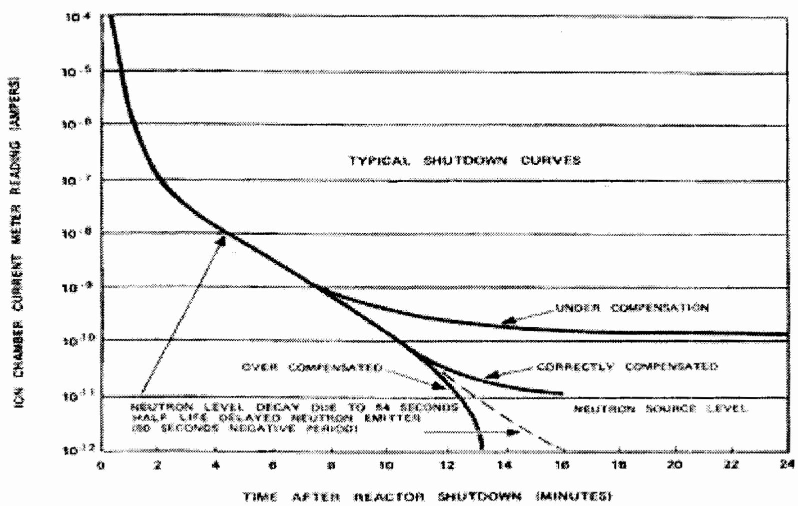
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.2	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the characteristics to identify the loss of a nuclear instrumentation detector.

High Cog: This question is high cog because the examinee must analyze the detector characteristics in the stem and determine which detector is correct.

FIGURE 4 - TYPICAL GAMMA COMPENSATED CURVE FOR A COMPENSATED ION CHAMBER



EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

58 **ID: 2084857** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Tave in each loop is as follows:

RCS Loop A Tave	580F
RCS Loop B Tave	578F
RCS Loop A&B Average	579F

- RCS Loop A&B is the selected Tave

Event:

- The Reactor Operator observes the following indications for RCS flow:
  - RCS loop A LOWERS to 33 x 10E6 lbm/hr
  - RCS loop B RISES to 78 x 10E6 lbm/hr

Which of the following describes the operation of RC-12-TaS, Tave Auto/Manual Transfer Switch, and what effect would this have on control rods?

Loop __(1)__ is selected automatically, and ICS would __(2)__ control rods.

A. (1) A
(2) insert

B. (1) B
(2) insert

C. (1) A
(2) withdraw

D. (1) B
(2) withdraw

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) RC-12-TaS is a transfer switch that selects the RCS loop with the highest flow (if one loop is greater than 95% flow and the other is less than 95%) for Tave control. This Tave goes to the console digital Tave to the ICS (TQ-TM-104-624-C001, Rev 5 Page 28). (2) The examinee must recognize that the Tave control will switch to the B loop. (3) Once the examinee has correctly determined the Tave that is controlling, they will have to look at the actual Tave itself and determine how will effect the ICS signal. Since the B loop Tave is lower than the RCS Loop A&B Average, this will create a positive neutron error which will withdraw control rod (TQ-TM-104-621-C001, Rev 10 Page 50).

A.	(1) A (2) insert	INCORRECT: (1) Plausible if the examinee believes that the loop with the lowest flow gets selected for Tave control. Incorrect because the loop with the highest flow gets selected. (2) Plausible if the examinee incorrectly determines the Tave effect on the Reactor control. On the console, the error will show up reverse as the ICS signal. Incorrect because this event will lead to a rod withdraw.
B.	(1) B (2) insert	INCORRECT: (1) Correct answer. (2) Plausible if the examinee incorrectly determines the Tave effect on the Reactor control. On the console, the error will show up reverse as the ICS signal. Incorrect because this event will lead to a rod withdraw.
C.	(1) A (2) withdraw	INCORRECT: (1) Plausible if the examinee believes that the loop with the lowest flow gets selected for Tave control. Incorrect because the loop with the highest flow gets selected (2) Correct answer.
D.	(1) B (2) withdraw	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	016		K4.03
	Importance Rating	2.8		

K/A: Nonnuclear Instrumentation: Knowledge of the NNIS design feature(s) and/or interlock(s) which provide for the following: Input to control system

Proposed Question:	Question #58	
Technical Reference(s):	TQ-TM-104-621-C001, Rev 10	TQ-TM-104-624-C001, Rev 5
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	624-GLO-5	

Question Source:	Bank #		
	Modified Bank #	371840	

	New			
Question History:	Unmod on Sys Exam 11	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how the Tave NNI system inputs into the ICS control system.

High Cog: This question is high cog because the examinee must analyze the current Tave and flow conditions and then determine the plant response.

356776
Plant Conditions:

- Operating at 100% power.
- ICS in full AUTOMATIC.

Event:

- The Reactor Operator observes the following indications for RCS flow:
  - RCS loop A LOWERS to 33 x 10^6 lbm/hr.
  - RCS loop B RISES to 78 x 10^6 lbm/hr.

Which of the following describes the operation of RC-12-TaS, Tave Auto/Manual Transfer Switch?

A. Automatically selects loop A.

B. Automatically selects loop B.

C. Allows the operator to select only loop A.

D. Allows the operator to select only loop B.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

59 **ID: 2105395** **Points: 1.00**

Sequence of Events:

- A LOCA has occurred
- Several ECCS component failures have occurred
- The crew has been directed to enter OP-TM-EOP-008, RCS SUPERHEATED
- The crew has determined that Fuel Clad Temperature has peaked at 1600°F, before HPI cooling was restored.
- Subsequently, the crew enters OP-TM-EOP-002, LOSS OF 25°F SUBCOOLING MARGIN.

Given the above information, determine:
(1) Whether the fuel has remained covered
(2) Whether the Hydrogen Purge and Recombiner are required for design accident mitigation

A. (1) At no time was any portion of the fuel uncovered
(2) Neither are required

B. (1) At no time was any portion of the fuel uncovered
(2) Both are required

C. (1) At least a portion of the fuel was uncovered at some time during this event
(2) Neither are required

D. (1) At least a portion of the fuel was uncovered at some time during this event
(2) Both are required

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-EOP-0081, RCS SUPERHEATED BASIS DOCUMENT (Rev 1, Page 3), superheat conditions can only occur when fuel is not covered with liquid. This will occur in the initial moments after a large LOCA but the fuel should be covered with liquid within 4 minutes of any design basis event. (2) Although hydrogen production would occur under such conditions, neither the Hydrogen Recombiner (1104-62, HYDROGEN RECOMBINER, REV 35 PAGE 4) or Hydrogen Purge System (OP-TM-823-902, POST ACCIDENT PURGE, Rev 3 Page1) are required to serve an accident mitigation function.

A.	(1) At no time was any portion of the fuel uncovered (2) Neither are required	INCORRECT: (1) Plausible because there is a further curve (Curve C) that could be exceeded in OP-TM-EOP-008, which the examinee may believe must be exceeded prior to fuel being uncovered. Incorrect because the basis document says when OP-TM-EOP-008 was entered fuel has already been uncovered. (2) Correct Answer.
B.	(1) At no time was any portion of the fuel uncovered (2) Both are required	INCORRECT: (1) Plausible because there is a further curve (Curve C) that could be exceeded in OP-TM-EOP-008, which the examinee may believe must be exceeded prior to fuel being uncovered. Incorrect because the basis document says when OP-TM-EOP-008 was entered fuel has already been uncovered (2) Plausible because both systems would be used if a casualty this severe would occur. Incorrect because neither system is required for design basis accident mitigation.
C.	(1) At least a portion of the fuel was uncovered at some time during this event (2) Neither are required	CORRECT: See above
D.	(1) At least a portion of the fuel was uncovered at some time during this event (2) Both are required	INCORRECT: (1) Correct answer. (2) Plausible because both systems would be used if a casualty this severe would occur. Incorrect because neither system is required for design basis accident mitigation.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	028		A1.01
	Importance Rating	3.4		

K/A: Hydrogen Recombiner and Purge Control: Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Hydrogen Concentration

Proposed Question:	Question #59	
Technical Reference(s):	OP-TM-823-902, Rev 3	OP-TM-EOP-0081
	1104-62, Rev 35	

Proposed References to be provided to applicants during examination:	None

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	EOP008-PCO-3		
Question Source:	Bank #		
	Modified Bank #	860274	
	New		
Question History:	unmod on Sim Exam 9	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that some hydrogen was released due to the peak fuel cladding temperature that was reached and that the Hydrogen Recombiner and Purge System are not required to mitigate a design basis accident. Operational considerations for casualties based on hydrogen concentration to not occur until SAMG space. The existance of hydrogen based on the RCS SUPERHEAT is why the hydrogen monitors are started.

Sequence of Events:

- A LOCA has occurred.
- Several ECCS component failures have occurred.
- The crew has been directed to enter OP-TM-EOP-008, RCS Superheated.
- The crew has determined that Fuel Clad Temperature has peaked at 1810°F, before HPI cooling was restored.

Given the above information, determine:
(1) Whether the fuel remained covered.
(2) The affects of the fuel during the event.

A. (1) At no time was any portion of the fuel uncovered;
(2) Some localized oxidation of the zirconium fuel cladding produced hydrogen.

B. (1) At no time was any portion of the fuel uncovered;
(2) Some localized oxidation of the zirconium fuel cladding produced hydrogen, and some degradation of cladding material occurred.

C. (1) At least a portion of the fuel was uncovered at some time during this event
(2) Some oxidation of the zirconium fuel cladding produced hydrogen, and some degradation of cladding material occurred.

D. (1) At least a portion of the fuel was uncovered at some time during this event;
(2) Some oxidation of the zirconium fuel cladding produced hydrogen, and significant degradation of cladding material occurred.

Answer D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

60 **ID: 2088045** **Points: 1.00**

How is the Reactor Building Purge Flow Rate controlled?

RB Purge Flow Rate is controlled by a combination of controlling air flow from the discharge of AH-E-6A/6B, Purge Supply Fans using __(1)__ and controlling air flow from the Air Intake Tunnel by controlling the position of __(2)__.

A. (1) AH-D-8B, RB Purge Manual Loader
(2) AH-D-82, Makeup Damper

B. (1) AH-D-8B, RB Purge Manual Loader
(2) AH-V-1A, Purge Exhaust Valve

C. (1) AH-D-82, Makeup damper
(2) AH-V-8B, RB Purge Manual Loader

D. (1) AH-D-82, Makeup damper
(2) AH-V-1A, Purge Exhaust Valve

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Air for the Reactor Building Purge is brought in by the AH-E-6A/B fans from outside into the reactor building via AH-V-1C and AH-V-1D. (2) Air is exhausted from the Reactor Building by the AH-E-7A/B fans via AH-V-1A and AH-V-1B. (302-831, Rev 58). (3) Between the AH-V-1A/B valves and the AH-E-7A/B fans is a filter (AH-F-1) which filters air from the Reactor Building and makeup from the Air Intake Tunnel via AH-D-82 (302-832 Rev 9). (4) When the operator adjusts the Manual Purge Loader in the control room, they are adjusting AH-D-8B. When they are setting AH-PC-1146 they are adjusting makeup air to the AH-E-7 suction via AH-D-82 (OP-TM-823-406, RB PURGE - CONTAINMENT CLOSED, Rev 12 Page 7).		
A.	(1) AH-D-8B RB Purge Manual Loader (2) AH-D-82 Makeup Damper	CORRECT: See above.
B.	(1) AH-D-8B RB Purge Manual Loader (2) AH-V-1A Purge Exhaust Valve	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that adjusting a purge exhaust valve will adjust air flow from the air intake tunnel.
C.	(1) AH-D-82 Makeup damper (2) AH-V-8B RB Purge Manual Loader	INCORRECT: (1) Plausible if he examinee believe makeup comes from outside via the AH-E-6. Incorrect because this is the Purge Manual Loader. (2) Plausible if the examinee
D.	(1) AH-D-82 Makeup damper (2) AH-D-1A Purge Exhaust Valve	INCORRECT: 1. Plausible if the examinee believes the Makeup Damper controller, controls the air flow form the Intermediate Building. 2. Plausible if the examinee believes that controlling AH-V-1A would adjust air flow form the Air Intake Tunnel

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029	A4.01
	Importance Rating	2.5	

K/A: Containment Purge: Ability to manually operate and/or monitor in the control room: Containment purge flow rate

Proposed Question:	Question #60	
Technical Reference(s):	OP-TM-823-406, Rev 12	302-831, Rev 58
	302-832, Rev 9	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	824-GLO-10	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how to manually control the containment purge in the control room.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

61 **ID: 2084859** **Points: 1.00**

Plant conditions:
- Plant startup in progress
- Reactor power is 10%
- Main Turbine Chest Warming is in progress
- FW-P-1B is in service
- Both Auxiliary Boilers are secured
- GS-V-4 (Main Steam supply to Gland Steam Regulator) is open

Event:
- The ICS selected B OTSG pressure fails to 1200 psig without a SASS actuation
- B OTSG Pressure is 850 psig and lowering quickly
- Main Turbine Header Pressure is 880 psig and lowering slowly

Based on the above conditions:

(1) Which OTSG is supplying steam to FW-P-1B?
(2) What actions will mitigate this transient?

A. (1) A OTSG
(2) Place ICS controller for “MS-V-3A/B/C or MS-V-4B” in hand and throttle

B. (1) A OTSG
(2) Place ICS controller for “MS-V-3A/B/C or MS-V-4B” in hand and throttle AND transfer MS-V-4A/B to backup loader

C. (1) B OTSG
(2) Place ICS controller for “MS-V-3A/B/C or MS-V-4B” in hand and throttle

D. (1) B OTSG
(2) Place ICS controller for “MS-V-3A/B/C or MS-V-4B” in hand and throttle AND transfer MS-V-4A/B to backup loader

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Steam is supplied to FW-P-1B from OTSG 'B' (302011, Rev 76). (2) When the selected ICS OTSG pressure fails high without a SASS actuation the Turbine Bypass Valves (TBVs) and Atmospheric Dump Valves (ADVs) will open to up lower pressure. (3) Operators must place the ICS controller MS-V-3A/B/C (TBVs) in hand and transfer the MS-V-4A (ADV) to the backup loader to regain control over OTSG pressure in accordance with OP-TM-411-451, MANUAL CONTROL OF TBVs/ADVs (Rev 7 Page 2).

A.	1) A OTSG (2) Place ICS controller for "MS-V-3A/B/C or MS-V-4B" in hand and throttle	INCORRECT: (1) Plausible if the examinee believes that FW-P-1B is on extraction steam. The examinee could believe that since OTSG 'A' pressure is higher than OTSG 'B' that steam to the extraction steam is coming from OTSG 'A' exclusively. Incorrect because steam for FW-P-1B is still coming from OTSG 'B' as extraction steam (the preferred source) is not at the correct pressure. (2) Plausible because the MS-V-4B can be controlled from the MS-V-3A/B/C controller under certain conditions. Incorrect because those conditions do not exist.
B.	(1) A OTSG (2) Place ICS controller for "MS-V-3A/B/C or MS-V-4B" in hand and throttle AND transfer MS-V-4A/B to backup loader	INCORRECT: (1) Plausible if the examinee believes that FW-P-1B is on extraction steam. The examinee could believe that since OTSG 'A' pressure is higher than OTSG 'B' that steam to the extraction steam is coming from OTSG 'A' exclusively. Incorrect because steam for FW-P-1B is still coming from OTSG 'B' as extraction steam (the preferred source) is not at the correct pressure. (2) Plausible because this would be the correct actions if this were to occur on a different pressure transmitter.
C.	(1) B OTSG (2) Place ICS controller for "MS-V-3A/B/C or MS-V-4B" in hand and throttle	INCORRECT: (1) Correct answer. (2) Plausible because the MS-V-4B can be controlled from the MS-V-3A/B/C controller under certain conditions. Incorrect because those conditions do not exist.
D.	(1) B OTSG (2) Place ICS controller for "MS-V-3A/B/C or MS-V-4B" in hand and throttle AND transfer MS-V-4A/B to backup loader	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041	K4.16
	Importance Rating	2.6	

K/A: Steam Dump/Turbine Bypass Control: Knowledge of SDS design feature(s) and/or interlock(s) which provide for the following: Low main steam pressure

Proposed Question:	Question 61	
Technical Reference(s):	TQ-TM-104-411-C001, Rev 8	TQ-TM-104-401-C001, Rev 11

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	302-011, Rev 76			OP-TM-411-451, Rev 7	
Proposed References to be provided to applicants during examination:					None
Learning Objective:	621-GLO-8				
Question Source:	Bank #	371786			
	Modified Bank #				
	New				
Question History:	N/A	Last NRC Exam:	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis		X		
10 CFR Part 55 Content:	55.41	b.7			
	55.43				

Comments:

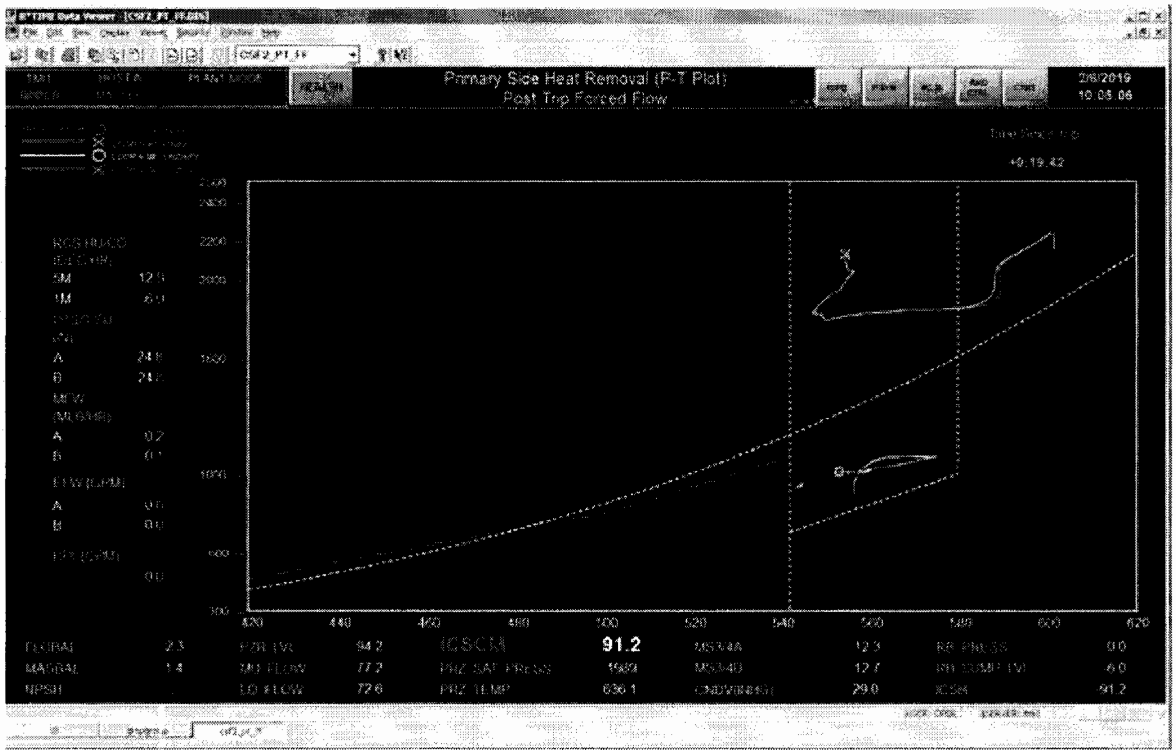
KA Match: This question matches the KA because the examinee must know the Main Steam Turbine Bypass Valve / Atmospheric Dump Valve ICS design feature that in Hand only the Turbine Bypass Valves are controlled from that controller. In addition, since the pressure fails high, the main steam pressure will be lowering until the operator takes both valve controlling stations to hand and throttles.

High Cog: This question is high cog because the examinee must know which OTSG will be supplying steam to FW-P-1B and how the controllers for the TBV/ADVs work.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

62 **ID: 2105461** **Points: 1.00**



Which of the following events will cause the P-T plot screen to have this shape and parameters?

A. Loss of Offsite Power

B. Loss of all Feedwater

C. Turbine Trip from 100% power

D. Unisolable Steam Line Rupture

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The image in the question is a P-T plot after a Turbine Trip from 100% power in from the simulator. (2) The shape of the curve is due to the cooldown and shrinkage of RCS water due to the trip. Initially Pressurizer level goes down until the cooldown subsides then pressure will start to rise to the normal operating bank. (3) In addition there is MFW flow indication on the left and side which the examinee can determine that there is feedwater flow.		
A.	Loss of Offsite Power	INCORRECT: Plausible because this will cause a Reactor and Turbine trip. Incorrect because the natural circulation box would come up and there would be no indication for MFW flow.
B.	Loss of all Feedwater	INCORRECT: Plausible because this would cause a Reactor and Turbine Trip. Incorrect because there is still MFW flow on the left hand of the plot.
C.	Turbine Trip from 100% Power	CORRECT: See above
D.	Unisolable Steam Line Rupture	INCORRECT: Plausible because an RPS setpoint would be reached and the crew would trip the Reactor. Incorrect because an Unisolable Steam Line Rupture would cause an excessive heat transfer which would drag the plot back further down past left side of the box.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045	A1.05
	Importance Rating	3.8	

K/A: Main Turbine Generator: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of primary plant parameters (temperature and pressure) following a T/G trip

Proposed Question:	Question 62	
Technical Reference(s):	Plant Simulator	
Proposed References to be provided to applicants during examination:		None
Learning Objective:	EOP001-PCO-2	
Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	N/A	Last NRC Exam: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know (and be able to monitor) the correct primary parameters (pressure and temperature) on a trip of the plant. Specifically the examinee must identify that the it was the turbine that tripped and not any of the other malfunctions.

High Cog: This question is high cog because the examinee will have to analyze the PT plot and parameters and identify the cause.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

63 **ID: 2105465** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

The Condensate and Feedwater lineup is as follows:

- CO-P-1A and CO-P-1B switches are in the NORM AFT START position
- CO-P-1C switch is in PULL TO LOCK position.
- CO-P-2A and CO-P-2B switches are in the NORM AFT START position.
- CO-P-2C switch is in NORM AFT STOP position.
- FW-P-1A is the last RESET feedpump.

Which of the following describes plant response to a trip of Condensate Pump, CO-P-1B?

A. Condensate Pump, CO-P-1C, will auto start

B. Condensate Booster Pump, CO-P-2C, will auto start

C. Condensate Booster Pump, CO-P-2A, will trip, only

D. Condensate Booster Pump, CO-P-2A, and Feedwater Pump, FW-P-1A will trip

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The Condensate, Condensate Booster Pumps, and Feedwater Pumps trip circuit involves using a counting circuit. (2) If the number of condensate pumps does not match number of booster pumps, the circuit will attempt to start a condensate pump. (3) CO-P-1C is in PTL and will not start. (3) After .5 seconds, if the counting circuit is still not satisfied, the circuit will sequentially trip booster pumps starting with CO-P-2A, then CO-P-1B until the counting circuit is satisfied. When two booster pumps are no longer running, the last reset FW pump will trip.

A.	Condensate Pump, CO-P-1C, will auto start	INCORRECT: Plausible because this is the normal plant response. Incorrect because CO-P-1C is in PTL.
B.	Condensate Booster Pump, CO-P-2C, will auto start	INCORRECT: Plausible if the examinee does not understand the current pump configuration.
C.	Condensate Booster Pump, CO-P-2A, is the only pump to trip	INCORRECT: Plausible if the examinee believes that only a condensate booster pump will trip and the counting circuit will not trip a Main Feedwater Pump.
D.	Condensate Booster Pump, CO-P-2A, and Feedwater Pump, FW-P-1A will trip	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	056	K1.03	
	Importance Rating	2.6		

K/A: Condensate: Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following: MFW

Proposed Question:	Question 63	
Technical Reference(s):	OP-TM-MAP-M0101, Rev 2	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	421-GLO-4	

Question Source:	Bank #	879453	
	Modified Bank #		
	New		

Question History:	System Exam 5	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the interrelation (cause-effect relationship) between condensate pumps and the feedwater pumps.

High Cog: This question is high cog because the examinee must analyze the abnormal lineup and determine the correct plant response.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

64 **ID: 2105468** **Points: 1.00**

Plant Conditions:

- "B" OTSG tube leak of approximately 20 gpm has occurred
- Plant Cooldown is in progress per OP-TM-EOP-005, OTSG Tube Leakage
- The most recent sample of RCS activity is 1.5 microcuries / ml
- B OTSG TURB BYP LINE RAD MONITOR, RM-G-27, is reading 2,000 cpm
- MAIN CONDENSER OFFGAS RAD MONITOR, RM-A-5, is reading 200 cpm

Event:

- At approximately 450 F, a small fuel failure occurs, only raising noble gas content in the RCS from 0.05 to 1.5 microcuries / ml

Given the above information and assuming only noble gasses are released from the fuel failure, RM-G-27 ___(1)___, and RM-A-5 ___(2)___.

A. (1) stays constant because noble gasses aren't seen by RM-G-27
(2) approximately doubles because RCS activity approximately doubles

B. (1) stays constant because noble gasses aren't seen by RM-G-27
(2) goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30

C. (1) approximately doubles because the total RCS activity approximately doubles
(2) approximately doubles because RCS activity approximately doubles

D. (1) approximately doubles because the total RCS activity approximately doubles
(2) goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) RM-G-27 measures activity in the steam line (TQ-TM-104-661-C001, Rev 7 Page 23). RM-G-27 counts double because the RCS activity roughly double (Most recent sample at 1.5 microcuries / ml + 1.5 microcuries / ml from the event). (2) RM-A-5 is the condenser off gas sampler. RM-A-5 goes up by a factor of 30 because the noble gas goes up by a factor of about 30. RM-A-5 mostly measures noble gasses which to stay in solution.		
A.	(1) stays constant because noble gasses aren't seen by RM-G-27 (2) approximately doubles because RCS activity approximately doubles	INCORRECT: Plausible if the examinee believes RM-G-27 stays constant because noble gasses aren't seen by RM-G-27, and RM-A-5 approximately doubles because RCS activity approximately doubles. Incorrect since RM-G-27 does measure all activity passing through the TBV steam lines.
B.	(1) stays constant because noble gasses aren't seen by RM-G-27 (2) goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30	INCORRECT: Plausible if the examinee believes RM-G-27 stays constant because noble gasses aren't seen by RM-G-27, and RM-A-5 goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30. Incorrect since RM-G-27 does measure all activity passing through the TBV steam lines.
C.	(1) approximately doubles because the total RCS activity approximately doubles (2) approximately doubles because RCS activity approximately doubles	INCORRECT: Plausible if the examinee believes RM-G-27 approximately doubles because RCS activity approximately doubles, and RM-A-5 approximately doubles because RCS activity approximately doubles. Incorrect since RM-A-5 is mostly measuring noble gasses which do not stay in solution.
D.	(1) approximately doubles because the total RCS activity approximately doubles (2) goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	072	K5.01	
	Importance Rating	2.7		

K/A: Area Radiation Monitoring: Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation theory, including sources, types, units, and effects

Proposed Question:	Question #64	
Technical Reference(s):	TQ-TM-104-661-C001, Rev 7	
Proposed References to be provided to applicants during examination:	None	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	661-GLO-7			
Question Source:	Bank #	469338		
	Modified Bank #			
	New			
Question History:	Simulator Exam 9	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	b.11		
	55.43			

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implications which may arise from the radiation monitor trend. The examinee must know radiation theory to understand the factors in which the radiation monitors rise due to a fuel failure.

High Cog: The question is high cog because the examinee must know and analyze the relationship between RCS activity and the radiation monitor readings.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

65 **ID: 2086971** **Points: 1.00**

Event:

- A malfunction has occurred such that all of the Ultraviolet (UV) Detectors associated with the Air Intake Tunnel Halon System are inoperable and incapable of automatically actuating the system.

Which of the following identifies how the Air Intake Tunnel Halon System has been affected by this malfunction?

The Air Intake Tunnel Halon System _______.

A. can be operated manually, ONLY

B. will be actuated automatically by the Accelerometer (AS40) actuating, ONLY

C. will be actuated automatically by any Pressure Wave (Explosion) Detector(s) actuating, ONLY

D. will be actuated automatically by either the Accelerometer (AS40) actuating or one Pressure Wave (Explosion) Detector actuating

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: Each Halon zone acts independently with its own ultraviolet (UV) and pressure wave detectors. There are 13 UV detectors, with a minimum of two detectors located in each zone. Additionally, there are 10 Pressure Wave Detectors, also with a minimum of two detectors in each zone. Any one ultraviolet detector or pressure wave detector will actuate the Air Intake Tunnel Halon System in the associated section. Consequently, the Air Intake Tunnel Halon System can be actuated automatically by one Pressure Wave (Explosion) Detector actuating. (TQ-TM-104-810-C001, Rev 11 Page 75)

A.	can be operated manually, ONLY	INCORRECT: Plausible because portions of the system, specifically the Grinnell Sprinkler Deluge System and the ventilation system re-alignment can be actuated manually. The operator may incorrectly believe that the system can only be actuated manually.
B.	will be actuated automatically by the Accelerometer (AS40) actuating, ONLY.	CORRECT: Plausible because the system has an Accelerometer. The system has one Accelerometer to detect airplane crashes into intake structure. This device provides for alarms in the Control Room, and the typical ventilation system responses, but will NOT actuate the Halon System. The operator may incorrectly believe that it is this device, and not the pressure wave detectors that will automatically actuate the system.
C.	will be actuated automatically by any Pressure Wave (Explosion) Detector(s) actuating ONLY	CORRECT: See above
D.	will be actuated automatically by either the Accelerometer (AS40) actuating or one Pressure Wave (Explosion) Detector actuating	INCORRECT: Plausible because the system has an Accelerometer. The system has one Accelerometer to detect airplane crashes into intake structure. This device provides for alarms in the Control Room, and the typical ventilation system responses, but will NOT actuate the Halon System. The operator may incorrectly believe that it is both this device, and the pressure wave detectors that will automatically actuate the system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086	K6.04
	Importance Rating	2.6	

K/A: Fire Protection System: Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: Fire, smoke, and heat detectors

Proposed Question:	Question #65	
Technical Reference(s):	TQ-TM-104-810-C001, Rev 11	
Proposed References to be provided to applicants during examination:		None
Learning Objective:	811-GLO-8	

EXAMINATION ANSWER KEY

Question Source:	Bank #	860305		
	Modified Bank #			
	New			
Question History:	System Exam 12	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	b.7		
	55.43			

Comments:

KA Match: This question matches the KA because the examinee must have knowledge on how a malfunction of the fire protection system effects the Air Intake Tunnel Halon System.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

66 **ID: 2105575** **Points: 1.00**

Which of the following elements/isotopes are checked in accordance with 1301-1 SHIFT AND DAILY CHECKS for Primary and Secondary Activity/Chemistry?

A. (1) Primary: Oxygen, Chlorides, Fluorides only
(2) Secondary: Iodine

B. (1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides
(2) Secondary: Iodine

C. (1) Primary: Oxygen, Chlorides, Fluorides only
(2) Secondary: Xenon

D. (1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides
(2) Secondary: Xenon

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In 1301-1 SHIFT AND DAILY CHECKS (Rev 178 page 35) operators check that Reactor Coolant and Secondary Chemistry is within the technical specification limits. The operators check Xenon, Iodine, Oxygen, Chlorides and Fluorides in the RCS. They check Iodine in the Secondary.		
A.	(1) Primary: Oxygen, Chlorides, Fluorides (2) Secondary: Iodine	INCORRECT: (1) Plausible because one section (D.3) checks for these elements. Incorrect because another section checks for Xenon and Iodine as well. (2) Correct answer.
B.	(1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides (2) Secondary: Iodine	CORRECT: See above
C.	(1) Primary: Oxygen, Chlorides, Fluorides (2) Secondary: Xenon	INCORRECT: (1) Plausible because one section (D.3) checks for these elements. Incorrect because another section checks for Xenon and Iodine as well (2) Plausible if the examinee believes we check for Xenon in the Secondary.
D.	(1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides (2) Secondary: Xenon	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes we check for Xenon in the Secondary.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.34	
	Importance Rating	2.7	

K/A: Knowledge of primary and secondary chemistry limits.

Proposed Question:	Question #66	
Technical Reference(s):	1301-1, Rev 178	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	551-GLO-6	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the elements/isotopes in the primary and secondary that have limits.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

67 **ID: 2105590** **Points: 1.00**

Plant conditions:

- A refueling outage has commenced
- The core is in the process of being off-loaded

Which ONE of the following describes the correct action if RM-G-9, FUEL HANDLING BUILDING FUEL HANDLING BRIDGE, radiation monitor loses power?

A. Cease all Spent Fuel Pool fuel movement until proper portable survey instruments are provided to monitor radiation levels

B. Spent Fuel Pool fuel movements may continue as long as RM-A-4, FHB VENT RADIATION MONITOR, remains operable

C. Spent Fuel Pool fuel movements may continue as long as AH-E-10, FHB SUPPLY FAN is secured and the FHB isolation dampers are closed

D. Cease all Spent Fuel Pool fuel movement until FHB isolation dampers have been verified closed AND RM-A-4 FHB VENT RADIATION MONITOR has been verified operable

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1505-1, FUEL AND CONTROL COMPONENT SHUFFLES (Rev 62 Page 7) step 5.1.2 says if any RM-G unit becomes inoperable, that portable survey instrumentation must be used having appropriate range and sensitivity to fully protect individuals involved in fuel handling operations until permanent instrumentation is returned to service.		
A.	Cease all spent fuel pool fuel movement until proper portable survey instruments are provided to monitor radiation levels	CORRECT: See above
B.	Spent Fuel Pool movements may continue as long as RM-A-4, FHB VENT RADIATION MONITOR, remains operable	INCORRECT: RM-A-4 monitors FHB exhaust flow for radiation and would provide warning of off-normal airborne conditions. Incorrect because RM-G-9 is specifically required by 1505-1.
C.	Spent Fuel Pool movements may continue as long as AH-E-10, FHB SUPPLY FAN is secured and the FHB isolation dampers are closed	INCORRECT: Plausible if it is believed that satisfying the interlock to RM-G-9 would be correct. Incorrect because another portable instrument is still required.
D.	Cease all spent fuel pool fuel movement until FHB isolation dampers have been verified open AND RM-A-4 FHB VENT RADIATION MONITOR has been verified operable	INCORRECT: Plausible if it was believed that ensuring RM-A-4 could provide replacement interlock function for RM-G-9. Incorrect because another portable instrument would still be required.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	2.1.36		
	Importance Rating	3.0		

K/A: Knowledge of procedures and limitations involved in core alterations.

Proposed Question:	Question #67	
Technical Reference(s):	1505-1, Rev 62	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	661-GLO-10	
Question Source:	Bank #	895238
	Modified Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New			
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.12	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the procedures and limitations with core refueling.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

68 **ID: 2087057** **Points: 1.00**

Given the following plant conditions:

- Core age is 345 EFPD
- Plant startup is in progress
- Xenon Free Startup
- The reactor was made critical 6 hours ago
- Plant power ascension is on hold at 55% power and is expected to remain at this power level for the next 5 hours
- Control rod index is 273%
- Boron concentrations are as follows:
  - RCS boron concentration is 1334 ppm boron
  - RCBT 'A' boron concentration is 27 ppm boron
  - RCBT 'B' boron concentration is 1331 ppm boron
  - RCBT 'C' boron concentration is 2052 ppm boron
  - BWST boron concentration is 2455 ppm boron

In order to maintain the current control rod index over the next 5 hours, RCS boron must be controlled by making up from the _______.

A. 'A' RCBT

B. 'B' RCBT

C. 'C' RCBT

D. BWST

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Xenon concentration will be building towards equilibrium values during the 5 hour hold which adds negative reactivity. (2) In order to maintain the current rod position, the RCS boron concentration will need to be reduced in order to add positive reactivity. (3) Water must be added from a boron concentration less than the current boron concentration to dilute. (4) OP-TM-211-455, FEED FROM A RCBT (Rev 30 covers feeding from the bleed tanks and the effects on control rod position.		
A.	'A' RCBT	CORRECT: See above
B.	'B' RCBT	INCORRECT: Plausible since 'B' RCBT is close to the current RCS Boron Concentration but still less than RCS Boron. The examinee may believe Xenon will have minimal affect and this concentration would maintain CR Index for the time of 5 hours.
C.	'C' RCBT	INCORRECT: Plausible if examinee does not understand the direction Xenon concentration will go in for this short period of time.
D.	BWST	INCORRECT: Plausible if examinee does not understand the magnitude and direction Xenon concentration will go and a need to raise boron concentration rapidly to maintain rod position.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.43	
	Importance Rating	4.1	

K/A: Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion

Proposed Question:	Question #68	
Technical Reference(s):	OP-TM-211-455, Rev 3	
Proposed References to be provided to applicants during examination:	NONE	
Learning Objective:	SOER 94-2 (V.I.11.06)	

Question Source:	Bank #	1588528	
	Modified Bank #		
	New		
Question History:	N/A	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to select the correct tank in order to choose the correct procedure.

High Cog: The examinee has to analyze the plan conditions and understand the reactivity feedback from Xenon and the procedure response for maintaining the control rod position.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

69 **ID: 2105974** **Points: 1.00**

In accordance with 1102-4, POWER OPERATIONS, which of the following personnel must be notified prior to lowering power for Turbine Valve Testing?

A. Transmission System Operator, only

B. Transmission System Operator and the Nuclear Duty Officer, only

C. Transmission System Operator, Nuclear Duty Officer, and Power Team

D. Transmission System Operator, Nuclear Duty Officer, and Generation Dispatch

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance 1102-4, POWER OPERATIONS, (Rev 137 Page 9) if power change is greater than 10 MWe then perform the following notify the TSO, NDO and Power Team. (2) Reactor power must be lowered to less than 90% on the hightest reading NI, which is greater than 10 MWe.		
A.	Transmission System Operator, only	INCORRECT: Plausible if examinee thinks this is the only person necessary to notify for a change in MW production.
B.	Transmission System Operator and the Nuclear Duty Officer, only	INCORRECT: Plausible if examinee thinks these are the only people necessary to notify.
C.	Transmission System Operator, Nuclear Duty Officer, and Power Team	CORRECT:
D.	Transmission System Operator, Nuclear Duty Officer, and Generation Dispatch	INCORRECT: Plausible if examinee believes Generation Dispatch must be notified.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.17	
	Importance Rating	2.6	

K/A: Knowledge of the process for managing maintenance activities during power operation, such as risk assessments, work prioritization, and coordination with the transmission operator.

Proposed Question:	Question #69
Technical Reference(s):	1102-4, Rev 137
Proposed References to be provided to applicants during examination:	None
Learning Objective:	GOP-004-PCO-2

Question Source:	Bank #	N/A
	Modified Bank #	N/A
	New	X

Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.10

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the correct communication requirements with respect to reducing electrical load, and who must be notified with respect to grid operation.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

70 **ID: 2087168** **Points: 1.00**

Plant conditions:

- Reactor power is 100%, with ICS in full automatic.
- Current Day/Time is Monday at 0800.
- Makeup System valve alignment is normal for power operations.
- Makeup Pump Status:

	MU-P-1A	MU-P-1B	MU-P-1C
Operational Condition	Standby	Operating	Standby
Power Supply		1E 4 KV Swgr.	
Additional Information	Back-up pump for seal injection	Supplying MU and RCP seal injection	
ES Selected	Yes	No	Yes

Event:

- At 0900 hours MU-P-1C is declared inoperable due to a cracked casing vent.

What actions must be taken to place the plant on a 30 day time clock, ONLY?

Note:

- MU-V-76A/B: MU-P-1B/C DISCHARGE HDR X-CONNECT VALVES.
- MU-V-77A/B: MU-P-1A/B DISCHARGE HDR X-CONNECT VALVES.
- MU-V-16C/D: H.P.I. CONTROL VALVE C/D

A. Start MU-P-1A and then Stop MU-P-1B

B. Start EG-Y-1B and keep it running until MU-P-1C is returned to service

C. Unlock and open MU-V-76A and MU-V-76B to ensure MU-P-1B can supply HPI via MU-V-16C and MU-V-16D

D. Close MU-V-77A and MU-V-77B, unlock and open MU-V-76A and MU-V-76B, then ES Select MU-P-1B on the 1E 4KV Bus

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) Two Makeup Pumps must be operable in their engineering safeguards mode from their independent ES busses. (Technical Specification 3.3.1.1.b AMD 289 Page 3-21) (2) If only one Makeup Pump is operable in the engineering safeguards mode then the plant must enter Technical Specification 3.3.2 (ECR 14-00208 Page 3-23) which allows for one pump to be out of service for maintenance or testing. This is a 72 hour technical specification. (3) To get the plant to a 30 day AP-1038 timeclock only, the crew must close MU-V-77A and 77B, then unlock and open MU-V-76A and 76B, then ES select and start MU-P-1B on the 1E 4KV bus.		
A.	Start MU-P-1A and then Stop MU-P-1B	INCORRECT: Plausible if the examinee because this would keep the 'A' train Makeup Pump supplying seal injection. Incorrect because the crew would still have to swap the discharge cross connects.
B.	Start EG-Y-1B and keep it running until MU-P-1C is returned to service	INCORRECT: Plausible because starting an Emergency Diesel generator and loading it on a bus will get the plant out of some restrictive technical specifications. Incorrect because it will do nothing for this technical specification.
C.	Open MU-V-76A and MU-V-76B to ensure MU-P-1B can supply HPI via MU-V-16C and MU-V-16D	INCORRECT: Plausible because these actions will allow MU-P-1B supply the 'B' ES train. Incorrect because they would need to complete the separation by closing the other side cross connects and ES selecting the MU-P-1B.
D.	Close MU-V-77A and MU-V-77B, unlock and open MU-V-76A and MU-V-76B, then ES Select MU-P-1B on the 1E 4KV Bus	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.42	
	Importance Rating	3.9	

K/A: Ability to recognize system parameters that are entry level conditions for Technical Specifications

Proposed Question:	Question #70	
Technical Reference(s):	Technical Specification 3.3	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	211-GLO-10	

Question Source:	Bank #	354988	
	Modified Bank #		
	New		

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.8
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know that the given plant condition requires entry into Tech Specs since there is only one Train of HPI Operable. In addition by taking the appropriate actions, the Tech Spec Time Clock would not be applicable.

High Cog: The examinee must analyze the plant conditions and determine the actions required to return to 2 Trains of HPI being Operable

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

71 **ID: 2087171** **Points: 1.00**

OP-AA-108-101, CONTROL OF EQUIPMENT AND SYSTEM STATUS, contains directions on the use of the Abnormal Component Position Sheet (ACPS).

The ACPS provides a controlled method for ________.

A. aligning equipment outside of routine operations.

B. changing component positions for tasks involving more than 1 system

C. temporarily changing positions of components that have an effect on the UFSAR

D. positioning components as part of a Temporary Configuration Change Package (TCCP)

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-AA-108-101, CONTROL OF EQUIPMENT AND SYSTEM STATUS (Rev 14 Page 4) Step 4.1.1.1 says to utilize an ACPS for aligning equipment outside of routine operations.		
A.	aligning equipment outside of routine operations	CORRECT: See above.
B.	changing component positions for task involving more than 1 system	INCORRECT: Plausible if the examinee believes that an ACPS would aid in completing work in a timely manner. Incorrect because step 4.1.2.2 prohibits use of an ACPS in this manner.
C.	temporarily changing positions of components that have an effect on the UFSAR	INCORRECT: Plausible if the examinee believes this method is acceptable because it is temporary. Incorrect because step 4.1.2.5 prohibits use of an ACPS in this manner.
D.	positioning components as part of a Temporary Configuration Change Package (TCCP)	INCORRECT: Plausible if the examinee believes this method could be used to document the changes required by a TCCP. Incorrect because step 4.1.2.1 prohibits use of an ACPS in this manner.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.14	
	Importance Rating	3.9	

K/A: Knowledge of the process for controlling equipment configuration change.

Proposed Question:	Question #71	
Technical Reference(s):	OP-AA-108-101, Rev 14	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EQC02018	

Question Source:	Bank #	375154	
	Modified Bank #		
	New		

Question History:	Simulator Exam 2	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the procedural requirements for changing equipment status that require an ACPS for configuration control.

High Cog:

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

72 **ID: 2106162** **Points: 1.00**

Using the picture above, when the monitor reaches the High Alarm Setpoint:
(1) What is the correct indication for the bar graph?
(2) What is the designed response of MAP C0101, RADIATION LEVEL HI?

	1 BAR GRAPH INDICATION	2 C0101 ALARM RESPONSE
A.	Orange	Reflash
B.	Orange	Does NOT Reflash
C.	Red	Reflash
D.	Red	Does NOT Reflash

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with TQ-TM-104-661-C001 (Rev 9 Slide 42) describes the orange as warning and red as high. (2) MAP C0101 will reflash when any radiation monitor comes into warning or alarm (Slide 29).		
A.	Orange Reflash	INCORRECT: (1) Plausible if examinee does not know the indication for a high alarm on a detector. (2) Correct answer.
B.	Orange Does NOT Reflash	INCORRECT: (1) Plausible if examinee does not know the indication for a high alarm on a detector. (2) Plausible because most alarms do not automatically reflash.
C.	Red Reflash	CORRECT: See above.
D.	Red Does NOT Reflash	INCORRECT: (1) Correct answer. (2) Plausible because most alarms do not automatically reflash.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.15	
	Importance Rating	2.9	

K/A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question:	Question #71	
Technical Reference(s):	TQ-TM-104-661-C001, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	661-GLO-5	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.12
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowlege a fixed instrument will respond to a high radiation level.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

73 **ID: 2085099** **Points: 1.00**

Which one of the following identifies a group of Post Accident Monitoring Instruments that are required to be OPERABLE by Technical Specification Table 3.5-3, Post Accident Monitoring Instrumentation?

A. Containment Pressure,
PORV Position Monitor, and
Steam Generator Pressure.

B. Containment Pressure,
PORV Position Monitor, and
RCS Cold Leg Temperature.

C. Containment Pressure,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

D. PORV Position Monitor,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with Technical Specification Table 3.5-3 (AMD 240 Page 3-40d) Containment Pressure, Steam Generator Pressure, and RCS Cold Leg Temperature are Post Accident Monitoring Instrumentation.		
A.	Containment Pressure, PORV Position Monitor, and Steam Generator Pressure.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.
B.	Containment Pressure, PORV Position Monitor, and RCS Cold Leg Temperature.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.
C.	Containment Pressure, Steam Generator Pressure, and RCS Cold Leg Temperature.	CORRECT: All instruments are listed in Table 3.5.3 of Tech Spec 3.5.5
D.	PORV Position Monitor, Steam Generator Pressure, and RCS Cold Leg Temperature.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.3	
	Importance Rating	3.7	

K/A: Ability to identify post-accident instrumentation

Proposed Question:	Question #73	
Technical Reference(s):	T.S. 3.5.5 Table 3.5-3, Amd 240	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	624-GLO-14	

Question Source:	Bank #	907843
	Modified Bank #	
	New	

Question History:	N/A	Last NRC Exam:	12-01

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know what post accident instrumentation is required by technical specifications.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

74 **ID: 2106063** **Points: 1.00**

Plant Conditions

- Reactor is at 100% power with ICS in full auto

Event:

- OTSG A level is slowly rising
- MAP J-1-5 OTSG A Level Hi Actuates
- LT-1044 OTSG level indicates 89%

In accordance with MAP J-1-5, the following manual actions must be taken:

A. Immediately Trip FW-P-1A and verify ICS runback to < 585 MWe

B. Take manual control of OTSG A Feedwater to maintain level < Alarm Setpoint

C. Immediately trip both Main Feedwater Pumps and go to OP-TM-EOP-001, REACTOR TRIP

D. Ensure Main Feedwater Isolation Valves FW-V-5A, FW-V-92A, FW-V-16A, and FW-V-17A are closed then go to OP-TM-EOP-001, REACTOR TRIP

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with MAP J (Rev 49 Page 7) the crew must verify the level on redundant diverse indication and then take manual control of OTSG A Feedwater as necessary to maintain level less than the alarm setpoint		
A.	Immediately Trip FW-P-1A and verify ICS runback to < 585 MWe	INCORRECT: Plausible because this would lower Feedwater flow and cause a plant runback. Incorrect because this is not the correct action at this time.
B.	Take manual control of OTSG A Feedwater to maintain level < Alarm Setpoint	CORRECT: See above.
C.	Immediately trip both Main Feedwater Pumps and go to OP-TM-EOP-001, REACTOR TRIP	INCORRECT: Plausible because tripping the plant be the correct action if the high level would have cause an HSPS isolation of Main Feedwater. Tripping the Main Feedwater Pumps would have been performed if levels were >97.5%. Incorrect because level did not exceed the HSPS Main Feedwater actuation setpoint.
D.	Ensure Main Feedwater Isolation Valves FW-V-5A, FW-V-92A, FW-V-16A, and FW-V-17A are closed then go to OP-TM-EOP-001, REACTOR TRIP	INCORRECT: Plausible because if level would have reached an HSPS Main Feedwater Isolation setpoint, these actions would have been taken. Incorrect because level not get that high so taking control of these valves and lowering level is the correct answer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	2.4.50		
	Importance Rating	4.2		

K/A: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual

Proposed Question:	Question #74	
Technical Reference(s):	MAP J, Rev 49	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	644-GLO-6	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the alarm setpoint is still in. The higher cognitive part of the question is to operate the controls associated with the alarm setpoint.

High Cog: This question is high cog because the examinee must identify an abnormal condition and determine the correct actions to take.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

75 **ID: 2105827** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- MU-P-1B is out of service
- MU-P-1A is supplying normal makeup and seal injection

Event:

- IC-V-3, Intermediate Cooling from Reactor Building Isolation Valve, fails closed
- Immediately following the closure of IC-V-3, the 1D 4160V bus trips

Following the trip of 1D 4160V Bus, the first procedure to enter is _______.

A. OP-TM-EOP-001, REACTOR TRIP

B. OP-TM-AOP-041, LOSS OF SEAL INJECTION

C. OP-TM-AOP-013, LOSS OF 1D 4160V BUS

D. OP-TM-AOP-032, LOSS OF INTERMEDIATE CLOSED COOLING

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When isolation valve IC-V-3 fails closed this will stop flow to ICCW cooled components in the reactor building. (2) One of the components cooled is the Reactor Coolant Pump thermal barriers. (3) When the 1D 4KV Bus is lost, Seal Injection flow is lost. (4) With no ICCW flow and no seal injection the reactor will trip. (5) When the reactor trips, the crew must enter OP-TM-EOP-001, REACTOR TRIP. (6) All other procedures will be entered after the reactor trip IMA's are complete.

A.	OP-TM-EOP-001, REACTOR TRIP	CORRECT: See above
B.	OP-TM-AOP-041, LOSS OF SEAL INJECTION	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.
C.	OP-TM-AOP-013, LOSS OF 1D 4160V BUS	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.
D.	OP-TM-AOP-032, LOSS OF INTERMEDIATE CLOSED COOLING	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.2	
	Importance Rating	4.5	

K/A: Knowledge of system setpoints, interlocks, and automatic actions associated with EOP entry conditions

Proposed Question:	Question #75	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	
Proposed References to be provided to applicants during examination:	NONE	
Learning Objective:	EOP-001-PCO-1	
Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know and interlock which leads to the EOP-001 entry.

High Cog: This question is high cog because the examinee must identify that a reactor trip occurred and the procedure that must be entered.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

76 **ID: 2095501** **Points: 1.00**

REFERENCE PROVIDED

Plant conditions:

- Refueling shutdown in progress
- DHR Train "A" aligned for RCS cooling
- Total Decay Heat Flow is 3000 gpm
- Incore thermocouple temperatures steady at 100 degrees F
- RCS water level is 30" above centerline

Event:

- A leak in the "A" Decay Heat train causes RCS Water Level to lower to 21" above centerline before being isolated.
- Incore thermocouple temperatures are now 115 degrees F, continuing to rise

Which selection below describes the actions the operators must take and the appropriate procedure to be implemented?

A. Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.

B. Place DH Train B in service IAW OP-TM-212-901, EMERGENCY DHR OPERATIONS, to lower incore temperatures.

C. Throttle DH Train A flow using OP-TM-212-451, CONTROL OF DH TRAIN A FLOW AND TEMPURATURES, to lower incore temperatures.

D. Place DH-P-1A in pull-to-lock and raise RCS water level in accordance with OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The entry criteria for OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL (Page 1, Rev 11) is exceeded when incore temperature is greater than 10F; (2) The examinee will have to determine that the RCS water level is below the DH pump limit of OP-TM-212-000, DECAY HEAT REMOVAL SYSTEM, Attachment 7.2 (Rev 24, Page 35) with the axis and regions blackened out. (3) OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP (Rev 5 Page 1) is used to fill makeup to the RCS from the Reactor Coolant Bleed Tanks when the Makeup System is in the shutdown mode.		
A.	Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.	INCORRECT: Plausible because stopping evolutions which are reducing RCS inventory is required by OP-TM-EOP-030 (Rev 11, Page 3). Incorrect because using EOP-010, Guide 9 is not the proper way to fulfill this step when the Makeup System is in the shutdown mode.
B.	Place DH Train B in service IAW OP-TM-212-901, EMERGENCY DHR OPERATIONS, to lower incore temperatures.	INCORRECT: Plausible because this would lower temperatures, which is the primary concern. Incorrect because the plant is operating at an RCS water level that is too low for DH pump operation for the current DH flow condition.
C.	Throttle DH-P-1A flow using OP-TM-212-451, CONTROL OF DH TRAIN A FLOW AND TEMPERATURES, to lower incore temperatures.	INCORRECT: Plausible because the examinee could believe that throttling the DH-P-1A flow to a higher flowrate will lower incore temperatures. Incorrect because the Decay Heat System is operating below the vortex curve and raising flow will cavitate DH-P-1A.
D.	Place DH-P-1A in pull-to-lock and raise RCS water level in accordance with OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP.	CORRECT ANSWER: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	025	2.1.25
	Importance Rating		4.2

K/A: Loss of Residual Heat Removal System: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question:	Question 76	
Technical Reference(s):	OP-TM-EOP-030, Rev 011	OP-TM-220-555, Rev 5
	OP-TM-212-000, Rev 24	
Proposed References to be provided to applicants during examination:	OP-TM-212-000, Attachment 7.2 with axis and regions blackened	
Learning Objective:	EOP030 - PCO-4	
Question Source:	Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	Modified Bank #	363675		
	New			
Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee will have to correctly interpret the figure to come up with the correct answer, and the question involves a loss of decay heat removal.

High Cog: This question is high cog because the examinee must analyze the question and correctly use the reference material to identify that the pump is running below its vortex limit.

SRO only: The question is SRO only because the examinee must know the content of the procedures in order to proceed to put the plant in a safe condition.

Initial plant conditions:

- Maintenance outage in progress.
- Reactor vessel head is removed
- DHR Train "A" aligned for RCS cooling.
- Incore thermocouple temperatures steady at 100 degrees F.

Event:

- Decay Heat Removal Pump discharge pressure and DHR system flow rate begin to oscillate excessively.
- Incore thermocouple temperatures are now 115 degrees F, continuing to rise.

Based on these conditions identify the ONE selection below that describes required operator actions, and the appropriate procedure to be implemented.

A. Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.

B. Reduce DH Train A flow using OP-TM-212-451, Control of DH Train A Flow and temperatures, until the flow and pressure oscillations stop.

C. Place DH Train B in service IAW OP-TM-212-901, Emergency DHR Operations, and then vent DH-P-1A using OP-TM-212-553, Vent of DH-P-1A.

D. Place DH-P-1A in pull-to-lock and evacuate all personnel from the Reactor Building (RB) by actuating the RB Evacuation alarm IAW EOP-030, Loss of Decay Heat Removal.

Answer	D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

77 **ID: 2095536** **Points: 1.00**

Plant Conditions:

- Plant is operating at 88% power with ICS in auto

Sequence of Events:

- NI-7 fails high
- During a troubleshooting an electrical transient has caused a loss of Vital Bus 'D' power.
- The URO reports that Reactor Power is 90% and steady

(1) Which one of the following actions must the CRS direct?
(2) If an Emergency Action Level (EAL) threshold has been met?

A. (1) trip the reactor due to an ATWS
(2) an EAL threshold has been met

B. (1) trip the reactor due to an ATWS
(2) an EAL threshold has NOT been met

C. (1) shutdown due to a violation of Technical Specification 3.5, OPERATIONAL SAFETY INSTRUMENTATION
(2) an EAL threshold has been met

D. (1) shutdown due to a violation of Technical Specification 3.5, OPERATIONAL SAFETY INSTRUMENTATION
(2) an EAL threshold has NOT been met

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) When NI-7 fails high, there will be a locked in trip signal. (2) When the loss of Vital Bus 'D' occurs, RPS channel D will lose power causing a separate trip signal. (3) With two trip signals locked in the examinee must identify that a reactor trip should have occurred; (4) When the trip does not occur an ATWS condition exists and the reactor operator must trip the reactor trip (OS-24, Section 5.5, Step 4, Rev 30, Page 25); (5) In accordance with EP-AA-1009 Addendum 3 (Rev 2, Page 2-5) there are 2 separate criteria for an Unusual Event: An automatic trip did not work, and the subsequent manual trip did work; or the manual trip did not work and the subsequent automatic trip did work. (5) For the question, the examinee must identify that the automatic trip did not work, which meets an EAL threshold.

A.	(1) trip the reactor due to an ATWS (2) an EAL threshold has been met	CORRECT ANSWER: See above.
B.	(1) trip the reactor due to an ATWS (2) an EAL threshold has NOT been met	INCORRECT: (1) Correct Answer. (2) Plausible due to the separate criteria of ATWS. The examinee may believe that ONLY an automatic trip ATWS may not meet the EAL threshold. Incorrect because it does meet the EAL threshold.
C.	(1) shutdown due to a violation of Technical Specification 3.5, OPERATIONAL SAFETY INSTRUMENTATION (2) an EAL threshold has been met	INCORRECT: (1) Plausible because the examinee could believe that the required action for not meeting the Technical Specification Requirements is a plant shutdown. Incorrect because OS-24 directs the reactor trip. (2) Correct Answer.
D.	(1) shutdown due to a violation of Technical Specification 3.5, OPERATIONAL SAFETY INSTRUMENTATION (2) an EAL threshold has NOT been met	INCORRECT: (1) Plausible because the examinee could believe that the required action for not meeting the Technical Specification Requirements is a plant shutdown. Incorrect because OS-24 directs the reactor trip. (2) Plausible due to the separate criteria of ATWS. The examinee may believe that ONLY an automatic trip ATWS may not meet the EAL threshold. Incorrect because it does meet the EAL threshold.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	029	2.4.41
	Importance Rating		4.6

K/A: Anticipated Transient Without Scram: Knowledge of emergency action level thresholds and classification.

Proposed Question:	Question 77	
Technical Reference(s):	OS-24, Rev 30	EP-AA-1009, Addendum 3, Rev 2
	OP-TM-MAP-G0302, Rev 0	
Proposed References to be provided to applicants during examination:	None	

Learning Objective:	EOP001-PCO-4			
Question Source:	Bank #			
	Modified Bank #	986736		
	New			
Question History:		Last NRC Exam:		Unmodified on 12-01

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee must identify that an ATWS has occurred and know if an EAL threshold has been met.

High Cog: This question is high cog because the examinee must analyze the conditions and identify that an ATWS has occurred and then determine the correct plant response.

SRO only: EAL thresholds are unique to the SRO position. They must assess the plant conditions and then have knowledge of the EAL procedure to know that an EAL threshold has been met.

Plant Conditions:

- The plant is operating at 88% power.

Sequence of Events:

- ICS is placed in Manual Control IAW OP-TM-621-471, ICS Manual Control.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Normal" position by an I&C Technician.
- A Manual Channel Reset was **NOT** performed on the "A" RPS cabinet.
- The Shutdown Bypass switch in the "B" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The URO reports that Reactor Power is 90% and steady.

Given the above information, and assuming no other manual actions occurred, the CRS will direct the crew to ___(1)__, and the Tech Spec basis is to ___(2)___.

A. (1) lower power IAW 1102-4, Power Operations, due to a Nuclear Overpower condition
(2) prevent damage to the fuel cladding from rapid reactivity excursions

B. (1) lower power IAW 1102-4, Power Operations, due to a Nuclear Overpower condition
(2) prevent normal operation with part of the reactor protection system bypassed

C. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip, due to an ATWS
(2) prevent damage to the fuel cladding from rapid reactivity excursions

D. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip, due to an ATWS
(2) prevent normal operation with part of the reactor protection system bypassed

Answer D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

78 **ID: 2095585** **Points: 1.00**

OP-TM-EOP-003, XHT is entered due to a unisolable steam line rupture of the 'A' OTSG.

Current Plant Parameters:

- Tave is 520^{0}F and lowering
- 'A' OTSG pressure is 150 psig and lowering
- RCS pressure is 1900 psig and lowering
- Cooldown rate is 90 ^{0}F/HR and lowering
- Tube-to-Shell Delta Temp - A OTSG (PPC Point C4015) is -20^{0}F
- Rule 3, XHT has been completed and Guide 9, RCS INVENTORY CONTROL has been initiated

Which of the following actions must be taken?

A. Minimize Subcooling Margin

B. Secure a Reactor Coolant Pump

C. Feed the 'B' OTSG with EFW at < 435 gpm

D. Open a MU-V-14 (MU Pump Suction from BWST)

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the candidate will have to know/assess the following conditions: (1) An unisolable steam line rupture will cause an uncontrollable cooldown of the RCS; (2) The operating crew will perform Rule 3, to attempt to isolate the OTSG and remove all feedwater sources from the affected OTSG; (3) Due the cooldown, the stem states that Tave falls below 525^{0}F, which in accordance with OP-TM-EOP-003 the operating crew would initiate to other procedures of OP-TM-EOP-010: Rule 5 (Rev 9, Page 3) and Rule 6; (4) Rule 5 provides actions for Tave falling below 525^{0}F regarding emergency boration, Rule 6 deals with Tavg falling below 525^{0}F which requires the crew to emergency borate using MU-V-14A or MU-V-14B (OP-TM-EOP-010, Rev 20, Page 11)		
A.	Minimize Subcooling Margin	INCORRECT ANSWER: Plausible because this procedure is initiated in the same step by Rule 6. Incorrect, although Rule 6 will be initiated, the rule only requires minimizing subcooling margin under specific criteria, none of which are met.
B.	Secure a Reactor Coolant Pump	INCORRECT ANSWER: Plausible because there is a temperature requirement to secure a reactor coolant pump in OP-TM-EOP-003. Incorrect because that is required when RCS Tcold 407F and all 4 reactor coolant pumps are on.
C.	Feed the 'B' OTSG with EFW at < 435 gpm	INCORRECT ANSWER: Plausible because there is a limit to feed an OTSG if it were dry AND primary to secondary heat transfer were not established. Incorrect because primary to secondary heat transfer still exists.
D.	Open the a MU-V-14 (MU Pump Suction from BWST)	CORRECT ANSWER: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	BWE05	2.4.21
	Importance Rating		4.7

K/A: Steam Line Rupture - Excessive Heat Transfer: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling, and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question:	Question 78, SRO Question 3	
Technical Reference(s):	OP-TM-EOP-003, Rev 9	OP-TM-EOP-0031, Rev 4
	OP-TM-EOP-010, Rev 20	
Proposed References to be provided to applicants during examination:		None
Learning Objective:	EOP DBIG PCO-2	
Question Source:	Bank #	
	Modified Bank #	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	New	X		
Question History:	None	Last NRC Exam:	None	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee must use parameters and logic to assess the stem to determine the crew must emergency borate (Reactivity Control Safety Function).

High Cog: This question is High Cog because the candidate must assess the plant conditions and determine temperature thresholds for performing Rule 5. The examinee must determine that opening MU-V-14A or MU-V-14B is the correct action to take.

SRO Only: The question requires the examinee to assess plant conditions and to know the content of the EOP and the supplemental procedure to know what action that must be taken.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

79 **ID: 2087714** **Points: 1.00**

Event:

- Station Blackout

Current Plant Conditions:

- Neither 1D or 1E 4160V busses have been restored
- EF-P-1 is running
- OTSG Pressures:

	PSIG
'A' OTSG	1010
'B' OTSG	1010

- RCS Temperatures:

	Thot (F)	Tcold (F)
Loop A	597	596
Loop B	543	542

- RCS Pressure is 1950 psig
- Incore temperature is 563F

Which of the following identifies the status of Primary to Secondary Heat transfer and the method of controlling RCS temperature?

Primary to Secondary Heat Transfer is ___(1)___; operators must ___(2)___.

A. (1) occuring
(2) throttle the MS-V-3's (Turbine Bypass Valves) to stabilize RCS temperature

B. (1) occuring
(2) throttle the MS-V-4's (Atmospheric Dump Valves) to stabilize RCS temperature

C. (1) NOT occurring
(2) open the MS-V-3's (Turbine Bypass Valves) to cooldown the RCS

D. (1) NOT occurring
(2) open the MS-V-4's (Atmospheric Dump Valves) to cooldown the RCS

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) During a Station Blackout, none of the 4kV busses are available; (2) The crew must enter OP-TM-EOP-012, STATION BLACKOUT (Rev 5), which will stabilize the plant using emergency feedwater and the atmospheric dump valves; (3) The crew must interpret plant conditions to ensure natural circulation is being established which is determined by using OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS, Guide 10, NATURAL CIRCULATION (Rev 10, Page 23); (4) The crew must determine that the requirements for natural circulation are NOT met (RCS Thot - Tcold is greater than 50F); (5) Based on the OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, definition of Primary to Secondary heat transfer cannot be confirmed without natural circulation; (6) To establish natural circulation, the crew must lower OTSG pressure using the available means, which would be using the Atmospheric Dump Valves (ADV's) which are air operated valves; (7) In accordance with Step 3.21 of OP-TM-EOP-012 (Rev 5, Page 11) the crew must feed with EFW and open the MS-V-4's (ADV's) to maximize the cooldown; (7) The MS-V-4's are still available because they have Two Hour Instrument Air as the motive force to move them (TQ-TQ-104-C001, Rev 6, Page 14).

A.	(1) occurring (2) open the MS-V-3's (Turbine Bypass Valves) to stabilize RCS temperature	Incorrect Answer: (1) Plausible because the examinee will have to assess whether heat transfer is adequate. Incorrect because it is not adequate. (2) Plausible because under normal circumstances, the MS-V-3's would be the normal valves to control OTSG pressure. Incorrect because these valves are latched closed due to condenser vacuum interlock (due to the Station Blackout).
B.	(1) occurring (2) open the MS-V-4's (Atmospheric Bypass Valves) to stabilize RCS temperature	Incorrect Answer: (1) Plausible because the examinee will have to assess whether heat transfer is adequate. Incorrect because it is not adequate. (2) Plausible because if primary to secondary heat transfer were adequate, the MS-V-4's would be used to stabilize RCS Temperature.
C.	(1) NOT occurring (2) open the MS-V-3's (Turbine Bypass Valves) to cooldown the RCS	Incorrect Answer: (1) Part 1 is correct. (2) Plausible because under normal circumstances, the MS-V-3's would be the normal valves to control OTSG pressure. Incorrect because these valves are latched closed due to condenser vacuum interlock (due to the Station Blackout).
D.	(1) NOT occurring (2) open the MS-V-4's (Atmospheric Dump Valves) to cooldown the RCS	Correct Answer: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	055		EA2.01
	Importance Rating			3.7

K/A: Station Blackout: Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system.

Proposed Question:	Question 79	
Technical Reference(s):	OP-TM-EOP-012, Rev 5	OS-24, Rev 30

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	OP-TM-EOP-010, Rev 20	TQ-TM-104-850-C001, Rev 6	
Proposed References to be provided to applicants during examination:			None
Learning Objective:	EOP-012-PCO-4		
Question Source:	Bank #	N/A	
	Modified Bank #	N/A	
	New	X	
Question History:	N/A	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must know the MS-V-4's (Atmospheric Dump Valves) availability still exists with a loss if the instrument air system during a Station Blackout. The MS-V-4's will still operate from the control room with power to the controllers being supplied from the Backup Loaders (powered from VBB), and the motive force to operate the valves being the Backup two hour air bottles. During a station blackout, no instrument air compressor is available to operate these valves.

High Cog: This question is high cog because the examinee must assess plant conditions and make a decision on how to establish primary to secondary heat transfer.

SRO Only: This question is SRO only because the examinee must assess the plant conditions and make a decision to enter an event-specific sub-procedure (i.e. Feedwater available and Primary Secondary Heat Transfer not being adequate) and determine to re-establish plant control.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

80 **ID: 2085628** **Points: 1.00**

Plant Conditions:

- Reactor Power is 100% with ICS in full auto
- MU-P-1C is supplying normal makeup/seal injection

Event:

- The following alarms have actuated simultaneously:
  - A-1-8 BATTERY 1B DISCHARGING
  - A-2-8 BATTERY CHARGER 1B/1D/1F TROUBLE
  - A-3-8 INVERTER 1B/1D/1F INVERTER SYSTEM TROUBLE
  - PRF 1-1-1 CRDM BRK TEST TROUBLE
  - NN-3-1 230 KV SUBSTATION TROUBLE
  - AA-3-2 7KV BUS TROUBLE
  - AA-3-3 4KV BOP BUS TROUBLE
  - AA-3-5 480V BOP BUS TROUBLE

Based on these conditions which selection below describes:
(1) The controlling procedure, and
(2) The required action

A. (1) OP-TM-AOP-023, "A" DC SYSTEM FAILURE
(2) Trip the Reactor

B. (1) OP-TM-AOP-023, "A" DC SYSTEM FAILURE
(2) Lower power until condensate flow is < 8.7 MLB/HR

C. (1) OP-TM-AOP-024, "B" DC SYSTEM FAILURE
(2) Trip the Reactor

D. (1) OP-TM-AOP-024, "B" DC SYSTEM FAILURE
(2) Lower power until condensate flow is < 8.7 MLB/HR

Answer: D

Answer Explanation

Explanation: To answer this question correctly the examinee must know: (1) The indications for a complete loss of the 'B' DC distribution panel (from the control room are): A-3-8, INVERTER 1B/1D/1F, AA-3-5, 480V BOP BUS TROUBLE, PRF1-1-1 CRD BREAKER TEST TROUBLE in accordance with OP-TM-AOP-024 'B' DC SYSTEM FAILURE (Page 1, Rev 7); (2) Any one of these individually or a combination of them without all three could indicate a degraded (lowering voltage) condition on the 'B' DC System; (3) If the complete loss of the 'B' DC distribution panel occurs the crew must lower power until condensate flow is less than 8.7 MLB/HR due to system recirculation valves opening up (OP-TM-AOP-0241, 'B' DC SYSTEM FAILURE BASIS DOCUMENT (Rev 8, Page 9).

A.	(1) OP-TM-AOP-023, "A" DC SYSTEM FAILURE (2) Trip the Reactor	INCORRECT: (1) Plausible because similar alarms come in for a loss of 'A' DC Distribution. (2) Plausible because if the examinee determined that it is a partial loss of 'B' DC, at 119V and lowering the crew must perform a reactor trip.
B.	(1) OP-TM-AOP-023, "A" DC SYSTEM FAILURE (2) Lower power until condensate flow is < 8.7 MLB/HR	INCORRECT: (1) Plausible because similar alarms come in for a loss of 'A' DC Distribution. (2) Correct Answer
C.	(1) OP-TM-AOP-024, "B" DC SYSTEM FAILURE (2) Trip the Reactor	INCORRECT: (1) Correct Answer (2) Plausible because if the examinee determined that it is a partial loss of 'B' DC, at 119V and lowering the crew must perform a reactor trip.
D.	(1) OP-TM-AOP-024, "B" DC SYSTEM FAILURE (2) Lower power until condensate flow is < 8.7 MLB/HR	CORRECT: See Above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	2.2.44
	Importance Rating		4.4

K/A: Loss of DC Power: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.

Proposed Question:	Question #80	
Technical Reference(s):	OP-TM-AOP-024, Rev 7	
	OP-TM-AOP-0241, Rev 8	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-024-PCO-4	

Question Source:	Bank #		
	Modified Bank #	375153	
	New		

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee must determine a complete loss of DC has occurred from control room indications. Once determined the examinee will have to determine how this loss will effect plant and system conditions by choosing to perform a reactor trip or plant shutdown to a specified power.

High Cog: This question is high cog because the examinee must analyze the control room alarms and then determine the correct course of action.

SRO Only: This question is SRO only because the examinee must assess plant conditions (alarms in the control room) to determine which power is lost, and the type of loss (i.e. that a complete loss of 'B' DC has occurred and not a partial loss which would lower voltage). Once that assessment is made, the examinee must determine the correct section of procedure to mitigate the abnormal condition.

Plant Conditions:

- 100% Rx power.
- MU-P-1C (Makeup Pump 1C) is supplying normal makeup/seal injection.

Event:

- The following alarms have actuated simultaneously:
  - A-1-8 Battery 1B Discharging.
  - A-2-8 Battery Charger 1B/1D/1F Trouble.
  - A-3-8 Inverter 1B/1D/1F Inverter System Trouble.
  - PRF 1-1-1 CRDM Brk Test Trouble.
  - NN-3-1 230 KV Substation Trouble (Loss of DCB)
  - AA-3-2 7 KV Bus Trouble.
  - AA-3-3 4 KV BOP Bus Trouble.
  - AA-3-5 480V BOP Bus Trouble.

Based on these conditions identify the ONE selection below that describes:
(1) The controlling procedure, and
(2) The required actions.

A. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Notify Auxiliary Operator to verify Makeup Pump, MU-P-1A, is ready for start.

B. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Ensure 1M DC Distribution Power Battery Select is in the "A" Position.

C. (1) Alarm response for AA-3-2, 7KV Bus Trouble.
(2) Notify Transmission System Operator (TSO) and trip the reactor.

D. (1) Alarm response for NN-3-1, 230 KV Substation Trouble.
(2) Notify Transmission System Operator (TSO) and trip the reactor.

Answer: B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

81 **ID: 2095587** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto

Sequence of Events:

- Grid voltage was fluctuating for two minutes and then stabilized at 228 KV
- The operators observe on SS-1 that ONLY the 1091 line is connected to TMI
- The Transmission System Operator has informed the CRS that loss of a major switchyard is causing the problem and there is no current estimate for restoration
- The TMI Main Generator is operating at the limits of the capability curve

Which action, if any, must the crew take in accordance with 1107-11, TMI GRID OPERATIONS?

A. No action required, all operability requirements of offsite sources are met

B. Start and load EG-Y-1A onto the 'D' 4160V Bus until all operability requirements for offsite sources are met

C. Declare both Auxiliary Transformers inoperable and enter TS 3.0.1 because grid voltage is insufficient to power safety-related ES loads

D. Start and load EG-Y-1A and EG-Y-1B and load them on their respective busses until all operability requirements for offsite sources are met

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Technical Specification 3.7.2.a.i (Amd 224, Page 3-42) requires two 230 kV lines to be in service. (2) Or in accordance with the exception 3.7.2e (Amd 278, Page 3-43) for only having one 230kV line continued reactor operation is permissible if one emergency diesel generator is started and run continuously until two lines have been restored. (3) In accordance with 1107-11 TMI GRID OPERATIONS, Section 3.4.8.2 (Rev 42, Page 34), the ES bus amperage is recorded, then the Emergency Diesel is started and loaded to 1 MW.

A.	No action required, all operability requirements of offsite sources are met	INCORRECT: Plausible because the examinee may believe that the 1091 line is sufficient. The 1091 and 1092 lines are capable of carrying 100% of TMI-1 generator output each. The examinee could believe that having one in service meets the requirements. Incorrect because Technical Specifications require two 230 kV lines to be in service or compensatory actions to be implemented.
B.	Start and load EG-Y-1A onto the 'D' 4160V Bus until all operability requirements for offsite sources are met	CORRECT: See above.
C.	Declare both Auxiliary Transformers inoperable and enter TS 3.0.1 because grid voltage is insufficient to power safety-related ES loads	INCORRECT: Plausible because if grid voltage had stabilized at < 223 kV this would have been the correct action (1107-11, Rev 42, page 18). Incorrect because voltage is not below 223 kV.
D.	Start and load EG-Y-1A and EG-Y-1B and load them on their respective busses until all operability requirements for offsite sources are met	INCORRECT: Plausible because the examinee could believe both emergency diesel generators must be started and loaded for stability of power to ES equipment. Incorrect because in accordance with 1107-11, only one emergency diesel generator is required to be started and loaded.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	077	AA2.09
	Importance Rating		3.9

K/A: Generator Voltage and Electric Grid Disturbances: Ability to determine and interpret the following as they apply to Generator Grid Disturbances: Operational status of emergency diesel generators

Proposed Question:	Question 81	
Technical Reference(s):	T.S 3.7, AMD 224	1107-11, Rev 42
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	740-GLO-14	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #	2049246	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	
	55.43	b.2

Comments:

KA Match: This question matches the KA because the examinee must know the operational status of an emergency diesel generator for a grid disturbance.

High Cog: This question is high cog because the examinee must assess the grid voltage and the amount of 230 kV lines and determine the required actions.

SRO Only: This question is SRO only because the examinee must assess grid conditions and apply the required actions to a technical specification.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

82 **ID: 2095592** **Points: 1.00**

A LOCA has occurred.

The Shift Emergency Director has declared FA1 due to a LOSS OF RCS.

All ECCS equipment has operated as designed.

What is the maximum time limit to notify the NRC Operations Center via the ENS line of the SAF1-1 DECLARATION OF EMERGENCY CLASSIFICATION in accordance with LS-AA-1020, EXELON REPORTABILITY MANUAL?

A. 15 minutes

B. 1 hour

C. 4 hours

D. 8 hours

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with LS-AA-1110, EXELON REPORTABILITY REFERENCE MANUAL - SAFETY (SAF) (Rev 27 Page 1) SAF1-1 that the licensee shall notify the NRC of a declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan. (2) In this case, the Emergency Plan has been activated by a Loss of RCS where all ECCS equipment has operated as designed. (3) The time limit (Page 2) to notify the NRC Operations center is immediately after the state and local notifications are made and not to exceed one hour via the ENS line.

A.	15 minutes	INCORRECT: Plausible because the licensee must make some notifications (State and Local) within 15 minutes. Incorrect because one hour is the limit.
B.	1 hour	CORRECT: See above.
C.	4 hours	INCORRECT: Plausible because some events require a 4 hour notification to the NRC. Incorrect because the time limit for this EAL declaration is one hour.
D.	8 hours	INCORRECT: Plausible because some events require an 8 hour notification to the NRC. Incorrect because the time limit for this EAL declaration is one hour.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	024	2.4.30
	Importance Rating		2.7

K/A: Emergency Boration: 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 transmission operator

Proposed Question:	Question #82	
Technical Reference(s):	LS-AA-1020, Rev 20	LS-AA-1110, Rev 27

Proposed References to be provided to applicants during examination: None

Learning Objective: ADM08005

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	N/A	Last NRC Exam:	N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	b.1

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of a notification event related to a situation where the ECCS system would have actuated. When the ECCS system actuates, it emergency borates the RCS.

SRO only: This question is SRO only because the examinee must know the reportability requirements to the NRC. This is special knowledge exclusive to the SRO position.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

83 **ID: 2095158** **Points: 1.00**

In accordance with 1507-7, FUEL TRANSFER SYSTEMS OPERATING INSTRUCTIONS, which ONE of the following describes a VIOLATION of refueling administrative requirements?

A. Fuel moves in the Spent Fuel Pool are being supervised by an individual with an inactive SRO license.

B. An irradiated fuel assembly has been transferred to the Spent Fuel Pool with both trains of ESF Ventilation secured.

C. The fuel grapple FULL DOWN position has been determined by ZZ tape reading rather than by the Digital Fuel Elevation reading.

D. The Main Fuel Bridge has been left unattended with the fuel grapple at GRAPPLE UP DISENGAGED and the bridge de-energized.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1505-7 (Rev 33, Page 4), irradiated fuel movements shall not be permitted in the Fuel Handling Building (FHB) unless an ESF Filtration System is in operation; (2) This administrative pre-caution is there to prevent violating Technical Specification 3.15.4 (Amd 3.15.4, Page 3-62e) which requires one train of ESF ventilation to be in service or moving fuel in the FHB is prohibited.		
A.	Fuel moves in the Spent Fuel Pool are being supervised by an individual with an inactive SRO license.	INCORRECT: Plausible because an active license applies for core geometry changes but not spent fuel movement in the FHB
B.	An irradiated fuel assembly has been transferred to the Spent Fuel Pool with both trains of ESF Ventilation secured.	CORRECT: See above.
C.	The fuel grapple FULL DOWN position has been determined by ZZ tape reading rather than by the Digital Fuel Elevation reading.	INCORRECT: Plausible because both are methods for determining grapple position but ZZ tape meets the administrative requirement.
D.	The Main Fuel Bridge has been left unattended with the fuel grapple at GRAPPLE UP DISENGAGED and the bridge de-energized.	INCORRECT: Plausible because properly securing the bridge is an administrative requirement. However, this is an acceptable method per 1507-3.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	036	AA2.02
	Importance Rating		4.1

K/A: Fuel-Handling Incident: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurence of a fuel handling incident.

Proposed Question:	Question 83	
Technical Reference(s):	T.S 3.15.4 Amd 278	
	1505-7, Rev 33	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EOP DBIG - PCO-5	

Question Source:	Bank #	575096	
	Modified Bank #		

	New		
Question History:	N/A	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	b.7

Comments:

KA Match: This question matches the KA because the examinee must analyze all of the question options and determine which one is a violation of procedure and technical specification 3.15.4.1.

SRO only: This question is SRO only because the Senior Operator is responsible to ensure the technical specifications are met prior to moving fuel.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

84 **ID: 2095594** **Points: 1.00**

Sequence of Events:

- A reactor trip from 100% power has occurred due to a loss of off-site power (LOOP)
- Emergency Feedwater CANNOT be established to either OTSG
- Incore thermocouples are indicating 592°F and rising slowly
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, has been entered
- SCM is 72°F and stable

Which ONE of the following identifies the actions that must be taken?

A. Go to OP-TM-EOP-009, HPI COOLING, and establish HPI/PORV cooling and perform Rule 1 if SCM is lost.

B. Go to OP-TM-EOP-009, HPI COOLING and go to OP-TM-EOP-002, LOSS OF 25^{O}F SUBCOOLING MARGIN, if SCM is lost.

C. Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and open the PORV and reclose it when SCM approaches 30oF.

D. Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and reduce OTSG Pressure so that secondary Tsat is 552-572 oF.

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) There are two major paths within OP-TM-EOP-004 based on the availability of feedwater. (2) For this question there is no feedwater available, on step 3.8 (Rev 12, Page 5) the crew will transition to OP-TM-EOP-009, HPI COOLING based on incore thermocouple temperature of 592°F and a SCM of 72°F correlating to a saturation pressure of 2445 psig. (3) Within OP-TM-EOP-009 (Rev 8, Page 3), if SCM is lost (< 25°F) the crew will perform Rule 1 (which can be performed from memory) then continue in OP-TM-EOP-009 without entering OP-TM-EOP-002, LOSS OF 25°F SUBCOOLING MARGIN.

A.	Go to OP-TM-EOP-009, HPI COOLING, and establish HPI/PORV cooling and perform Rule 1 if SCM is lost	CORRECT: See above.
B.	Go to OP-TM-EOP-009, HPI COOLING and go to OP-TM-EOP-002, LOSS OF 25°F SUBCOOLING MARGIN, if SCM is lost	INCORRECT: Plausible because the entry criteria for OP-TM-EOP-002 is a subcooling margin less than 25°F. Incorrect because that procedure is written to combat RCS leaks, which is not occurring in this question.
C.	Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and open the PORV and reclose it when SCM approaches 30°F	INCORRECT: Plausible because this is a step in OP-TM-EOP-004. Incorrect because no feedwater is available and the crew must go to OP-TM-EOP-009.
D.	Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and reduce OTSG Pressure so that secondary Tsat is 552-572 °F	INCORRECT: Plausible because this is a step in OP-TM-EOP-004. Incorrect because no feedwater is available and the crew must go to OP-TM-EOP-009.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	K/A #	074		2.2.49
	Importance Rating			4.6

K/A: Inadequate Core Cooling: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question:	Question 84	
Technical Reference(s):	OP-TM-EOP-004, Rev 12	OP-TM-EOP-009, Rev 8

Proposed References to be provided to applicants during examination:	None

EXAMINATION ANSWER KEY

Learning Objective:	EOP004-PCO-4				
Question Source:	Bank #	860137			
	Modified Bank #				
	New				
Question History:	System Exam 8	Last NRC Exam:	N/A		
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41				
	55.43	b.5			

Comments:

KA Match: This question matches the KA because Rule 1 requires the Reactor Coolant Pumps to be secured within 1 minute of Subcooling Margin and the crew has the ability to perform the rule with no reference. Rule 1 secures all running Reactor Coolant Pumps, initiates ES, and EFW.

High Cog: This question is high cog because the examinee must analyze plant conditions and determine that OP-TM-EOP-009 must be entered to establish HPI/PORV cooling.

SRO Only: This question is SRO only because the examinee must assess plant conditions and know the content of procedures in order to select a required course of action.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

85 **ID: 2094444** **Points: 1.00**

Plant Conditions:

- The reactor is at 83% power

Event:

- ICS RUNBACK alarm actuates
- Loop A main feedwater flow rapidly RISING
- Loop B main feedwater flow rapidly LOWERING
- Condensate flow indication is LOWERING

Which of the following procedures provides the required actions that mitigate these plant conditions?

A. OP-TM-MAP-H0101, ICS RUNBACK and 1102-4, POWER OPERATIONS

B. OP-TM-MAP-H0101, ICS RUNBACK and OP-TM-AOP-062, INOPERABLE ROD

C. OP-TM-AOP-010, LOSS OF THE 1A 4160V BUS and 1102-4, POWER OPERATIONS

D. OP-TM-MAP-K0106, POWER LOAD UNBALANCE and OP-TM-AOP-022, LOAD REJECTION

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) There are multiple conditions which will bring in the ICS RUNBACK alarm: Loss of a Reactor Coolant Pump (RCP), Loss of a Main Feedwater Pump, Asymmetric Rod fault, or RC Flow limit (OP-TM-MAP-H0101, Rev 2, Page 1); (2) Each abnormal condition causes a distinctly different Integrated Control System (ICS) response, for this condition an RCP in the 'B' RCS loop trips which causes the ICS to re-ratio Main Feedwater and the plant to runback to 665 MWe. For a RCP trip the ICS will raise FW in the loop with two running pumps, and lower the FW flow in the loop with one running RCP to maintain the primary to secondary heat balance all while running the plant back to the appropriate setpoint(TQ-TM-104-621-C001, Rev 10, Page 139-140); (3) After the examinee determines the cause of the abnormal condition, they will have to select the appropriate procedure to enter and mitigate the malfunction. For this specific malfunction, the crew must enter the ICS RUNBACK alarm response (OP-TM-MAP-H0101) and then determine that 1102-4, POWER OPERATION procedure includes crucial steps on how to operate the plant while it runs back.		
A.	OP-TM-MAP-H0101, ICS RUNBACK and 1102-4, POWER OPERATIONS	CORRECT ANSWER: See Above.
B.	OP-TM-MAP-H0101, ICS RUNBACK and OP-TM-AOP-062, INOPERABLE ROD	INCORRECT: Plausible because these would be the procedures entered if a control rod had dropped into the core. In addition, if a dropped rod were to occur then condensate flow would lower and the ICS RUNBACK alarm would come in as it does in the stem of the question.
C.	OP-TM-AOP-010, LOSS OF THE 1A 4160V BUS and 1102-4, POWER OPERATIONS	INCORRECT: Plausible because these procedures would be entered if we lost the 1A 4160V Bus. A loss of the 1A 4160V bus would cause a Main Feedwater pump and a string of condensate to trip, also causing the ICS RUNBACK alarm and condensate flow to lower while the plant was running back. Incorrect because FW would not re-ratio.
D.	OP-TM-MAP-K0106, POWER LOAD UNBALANCE and OP-TM-AOP-022, LOAD REJECTION	INCORRECT: Plausible because these procedures would be entered if the examinee believes the stem indicates a power load unbalance (PLU). A PLU would cause condensate flow to lower, but not the ICS RUNBACK Alarm or FW re-ratio.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	A01	AA2.1
	Importance Rating		3.7

K/A: Plant Runback: Ability to determine and interpret the following as they apply to the (Plant Runback): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:	Question #85	
Technical Reference(s):	OP-TM-MAP-H0101, Rev 2	
	TQ-TM-104-621-C001, Rev 10	
Proposed References to be provided to applicants during examination:	None	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Learning Objective:	621-GLO-11		
Question Source:	Bank #		
	Modified Bank #	357047	
	New		
Question History:	N/A	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments:

KA Match: This question matches the KA because the examinee must interpret the plant condition that caused the runback and select the appropriate procedure to mitigate the abnormal condition.

High Cog: This question is high cog because the examinee must assess the plant conditions, determine what the failed component is, and then select the correct procedure to mitigate the abnormal condition.

SRO Only: The question is SRO only because it requires the examinee to assess plant conditions and to know the content of the procedures in order to select a required course of action.

Plant Conditions:

- The reactor is at 83% power.

Event:

- ICS Runback alarm actuates.
- Loop A main feedwater flow rapidly RISING.
- Loop B main feedwater flow rapidly LOWERING.
- Condensate flow indication is LOWERING.

Based on these conditions, which one of the following equipment/component faults has occurred?

A. RC-P-1A tripped.

B. RC-P-1C tripped.

C. CO-P-2A tripped.

D. FW-P-1A tripped.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

86 **ID: 2107501** **Points: 1.00**

Plant Conditions:

- The reactor is operating at 100% power with ICS in full auto
- The following maintenance activities are in progress:
  - Repair of the air operator for sample isolation valve, CA-V-5A
    - CA-V-5A (Steam Generator A Feed Water Sample Valve) is closed
    - CA-V-4A (Steam Generator A Feed Water Sample Valve) is closed
  - Replacement of MU-V-16B HPI flow instrument, MU-FI-1127

Given the above information, which one of the following statements identifies:
(1) The maintenance that requires entry into a Tech Spec LCO, and
(2) The appropriate required time clock?

A. (1) CA-V-5A air operator repair
(2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within 6 hours

B. (1) CA-V-5A air operator repair
(2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours

C. (1) HPI flow instrument replacement
(2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within 6 hours

D. (1) HPI flow instrument replacement
(2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-211-000, MAKEUP AND PURIFICATION SYSTEM (Rev 36 Page 15), HPI flow indication (MU-FI-1126, 1127, 1128, & 1129) is required to be operable for the associated HPI train to be operable if that HPI train is also lined up to provide seal injection. When an HPI flow indicator is not operable, ensure compliance with Technical Specification 3.3.2 and initiate a 72 hr TS time clock. (2) The MU pump discharge cross connect valve lineup may be swapped to align an inoperable HPI flow indicator with the HPI train not lined up to seal injection. An inoperable HPI flow indicator associated with an HPI train not lined up to provide seal injection does not affect HPI operability per Tech Spec 3.3

A.	(1) CA-V-5A air operator repair (2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within 6 hours	INCORRECT: (1) Plausible because CA-V-5A is a containment isolation valve. Incorrect because CA-V-4A and CA-V-5A are closed which is the required position for containment isolation. (2) Plausible because this is a common Tech Spec time clock for many failed components. Incorrect because it does not apply to those component in this position.
B.	(1) CA-V-5A air operator repair (2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours	INCORRECT: (1) Plausible because CA-V-5A is a containment isolation valve. Incorrect because CA-V-4A and CA-V-5A are closed which is the required position for containment isolation. (2) Plausible because this is the time clock that would be in effect if both CA-V-4A and CA-V-5A were open.
C.	(1) HPI flow instrument replacement (2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within 6 hours	CORRECT: See above
D.	(1) HPI flow instrument replacement (2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours	INCORRECT: (1) Correct answer. (2) Plausible because this is the tech spec time clock if CA-V-4A and CA-V-5A were open.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	004	2.2.40
	Importance Rating		4.7

K/A: Chemical and Volume Control: Ability to apply Technical Specifications for a system

Proposed Question:	Question #86
Technical Reference(s):	OP-TM-211-000, Rev 36

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Proposed References to be provided to applicants during examination:				None
Learning Objective:	211-GLO-4			
Question Source:	Bank #	1685038		
	Modified Bank #			
	New			
Question History:	System Exam 14	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41			
	55.43	b.2		

Comments:

KA Match: This question matches the KA because the examinee must have the ability to apply a technical specification of the Makeup System.

SRO only: This question is SRO only because the examinee must identify a condition that places the plant in a 72 hour technical specificaton time clock.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

87 **ID: 2095067** **Points: 1.00**

Plant Conditions:

- Reactor is 100% with ICS in full auto
- A drought and heat wave have resulted in rising river water temperature and lowering river water level
- River water temperature is 91F and rising

The following alarms come in:

- MAP C-1-4 IC CRD FILTER DP HI
- MAP C-1-3 IC CRD CLG OUTLET TEMP HI

Current Intermediate Closed Cooling Water (ICCW) parameters are:

- IC-10 FI CRD FLOW is 105 GPM and steady
- IC-9 TI TEMP FROM CRD reached 160F and is now lowering

Identify (1) the actions the CRS must direct, and (2) the controlling plant procedure.

A. (1) Ensure MU-V-1A/B (Letdown Cooler Inlet Valves) are closed
(2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI

B. (1) Ensure MU-V-1A/B (Letdown Cooler Inlet Valves) are closed
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

C. (1) Cross-tie secondary river with nuclear river water
(2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI

D. (1) Cross-tie secondary river with nuclear river water
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Due to the elevated river water temperature, all closed cooling water systems that use the Susquehanna River as a heat sink will be operating at an elevated temperature; (2) The examinee should identify an Intermediate Closed flow issue to the control rod drives when the 2 alarms come in; (3) With the given parameters and the alarms, OP-TM-MAP-C0103 (Rev 3, Page 1) will be the only procedure in which entry conditions were met; (4) On IC-9, TEMP FROM CRD, temperature is lowering due to the MU-V-1A/B interlock that isolates the letdown coolers, which is a major heat load on the ICCW system.		
A.	(1) Ensure MU-V-1A/B (Letdown Cooler Inlet Valves) are closed (2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI	CORRECT: See above
B.	(1) Cross-tie secondary river with nuclear river water (2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI	INCORRECT: (1) Correct Answer. (2) Plausible because the examinee could believe that OP-TM-AOP-032 entry criteria have been met and actions must be given from that procedure.
C.	(1) Cross-tie secondary river with nuclear river water (2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI	CORRECT: (1) Plausible because cross tying secondary river with nuclear river water would give more cooling to ICCW components, incorrect because the criteria are not met to perform that procedure. (2) Correct answer.
D.	(1) Cross-tie secondary river with nuclear river water (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	INCORRECT: (1) Plausible because cross tying secondary river with nuclear river water would give more cooling to ICCW components, incorrect because the criteria are not met to perform that procedure. (2) Plausible because the examinee could believe that OP-TM-AOP-032 entry criteria have been met and actions must be given from that procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	008	A2.03
	Importance Rating		3.2

K/A: Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (2) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/Low CCW temperature.

Proposed Question:	Question #87	
Technical Reference(s):	OP-TM-MAP-C0103, Rev 3	
	OP-TM-AOP-032, Rev 5	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-032-PCO-4	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee has to predict that when letdown is isolated by the ICCW malfunction that pressurizer level will rise and based on that prediction the alarm response OP-TM-MAP-C0103 gives the direction ensure letdown is isolated.

High Cog: This question is high cog because the examinee must know why intermediate closed cooling water peaked and 160F and is now lowering, and that in response to that, the overall plant response.

SRO Only: This question is SRO only because it requires the examinee to assess the plant conditions and know the content of procedures in order to select a required course of action. This correct answer is not a major AOP, and more than entry criteria and overall mitigation strategy is required to be known to answer the question correctly.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

88 **ID: 2095203** **Points: 1.00**

Sequence of Events:

- An In-service Test (IST) performed on AH-E-1A, Reactor Building Air Handling Unit 1A, showed an air flow rate of 24530 cfm, which is less than the minimum flow rate allowed by the IST Acceptance Criteria.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Fan performance appeared to have leveled out at the lower flow rate, but still greater than the design minimum ASME value.
  - Engineering evaluations concluded that there would be no further degradation of flow rate.

(1) AH-E-1 must be declared __(1)__.
(2) What, if any, Technical Specification time clock must be entered?

A. (1) Degraded but Operable
(2) No Technical Specification time clock.

B. (1) Degraded but Operable
(2) 30 day AP 1038 timeclock for Three Train Safe Shutdown Systems Degraded

C. (1) Inoperable
(2) No Technical Specification time clock.

D. (1) Inoperable
(2) One RB Cooling fan can be removed from service for 7 days.

Answer: D

Answer Explanation

To answer this question correctly, the examinee must know: (1) From OP-AA-108-104, TECHNICAL SPECIFICATION COMPLIANCE, the definition of INOPERABLE is when an SSC is considered to be INOPERABLE when it is not capable of meeting all of the requirements of the Technical Specification, ATR, TRM, ISFSI, or ODCM definition for OPERABILITY; (2) Even though Engineering has determined that the pump will not degrade further, and that the flow is greater than the ASME value, this does not make the pump anything other than INOPERABLE because it is still lower than the minimum flow rate allowed by technical specifications. (2) Technical Specification 3.3.2 allows one train of the following systems to be out of service for 72 hours: Makeup and Purification, Decay Heat, RB Emergency Cooling, RB Spray, BWST level instrumentation, or cooling water systems for 72 hours; one of the exceptions to this is that one reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days (Technical Specification 3.3.3)

A.	(1) Degraded but Operable (2) No Technical Specification time clock.	INCORRECT: (1) Plausible because the term degraded only applies when the equipment is OPERABLE. The examinee may believe that the equipment is not INOPERABLE because the SSC can still meet ASME value, and will not degrade further, but since is below that technical specification value that the SSC is degraded. (2) Plausible because no technical specification would be entered if the equipment was degraded but operable.
B.	(1) Degraded but Operable (2) 30 day AP 1038 timeclock for Three Train Safe Shutdown Systems Degraded	INCORRECT: (1) Plausible because the term degraded only applies when the equipment is OPERABLE. The examinee may believe that the equipment is not INOPERABLE because the SSC can still meet ASME value, and will not degrade further, but since is below that technical specification value that the SSC is degraded. (2) Plausible because some systems have a 30 day AP 1038 timeclock. Incorrect because Reactor Building Ventilation is not one of those systems.
C.	(1) Inoperable (2) One RB Cooling fan can be removed from service for 7 days.	CORRECT: See above
D.	(1) Inoperable (2) One RB Cooling fan can be removed from service for 72 hours.	INCORRECT: (1) Correct Answer. (2) Plausible because if more than one train of RB emergency cooling were out of service, the plant would be in a 72 timeclock.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	022	2.2.12	
	Importance Rating			4.1

K/A: Containment Cooling: Knowledge of surveillance procedures

Proposed Question:	Question #88	
Technical Reference(s):	Technical Specification 3.3.2 and 3.3.3, ECR 14-00208	
	OP-AA-108-104, Rev 2	

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Proposed References to be provided to applicants during examination:					None
Learning Objective:	108104-APCO-1				
Question Source:	Bank #				
	Modified Bank #	1700778			
	New				
Question History:		Last NRC Exam:		Unmodified on 16-01	
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41				
	55.43	b.2			

Comments:

KA Match: This question matches the KA because the examinee must make the determination that AH-E-1A is inoperable even though the flow rate is greater than the ASME code and engineer has determined no further degradation will occur. In addition the examinee must know the AH-E-1 exception to Technical Specification 3.3.2 for when one train of the system is out.

High Cog: This question is high cog because the examinee must analyze the surveillance data and determine the status of AH-E-1A. In addition, the examinee must apply Technical Specification 3.3.2 and 3.3.3.

SRO only: This question is SRO only because the examinee must apply the Technical Specification required actions and understand the surveillance requirements of the system.

Sequence of Events:

- An In-service Test (IST) performed on Nuclear River (NR) system pump NR-P-1A showed a flow rate of 6234 gpm, which is less than the minimum flow rate allowed by Technical Specification.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Pump performance appeared to have leveled out at the lower flow rate, but still greater than the design minimum ASME value.
  - Engineering evaluations concluded that there would be no further degradation of flow rate.

After the troubleshooting plan is complete, NR-P-1A must be declared __(2)__.

A. Operable

B. Degraded

C. Inoperable

D. Unavailable

Answer: C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

89 **ID: 2088339** **Points: 1.00**

Plant Conditions:

- OTSG Tube Leakage exists in both the "A" and "B" OTSGs.
- OP-TM-EOP-001, Reactor Trip IMA's have been completed.
- The crew transitions to OP-TM-EOP-005, OTSG TUBE LEAKAGE.
- Radiation Monitors read as follows:
  - RM-A-5/15, Condenser Vacuum Pump Exhaust: 5.9 E3 CPM, in Hi Alarm, rising.
  - RM-A-5 HI, Condenser Vacuum Pump Exhaust: 3.7 E3 CPM, in Alert, rising.
  - RM-G-26, Main Steam Line From OTSG "A": 4.4 E3 CPM, in Hi Alarm, rising.
  - RM-G-27, Main Steam Line From OTSG "B": 5.6 E4 CPM, in Hi Alarm, rising.
- OTSG "A": Pressure is 1010 psig and steady; Level is 35% and steady.
- OTSG "B" Pressure is 985 psig and steady; Level is 50% and steady.
- RCS Pressure is 980 psig and lowering.

Event:

- A Loss of Offsite Power (LOOP) occurs.
- Offsite dose assessor reports that projected offsite integrated dose is 500 mRem TEDE.

(1) Which OTSG(s) must be isolated?
(2) How will cooling be provided to the RCS?

A. (1) Isolate OTSG 'B', only
(2) Throttle open MS-V-4A, OTSG 'A' Atmospheric Dump Valve, in accordance with Guide 12, RCS STABILIZATION

B. (1) Isolate OTSG 'B', only
(2) Throttle open MS-V-3D/E/F, OTSG 'B' Turbine Bypass Valves, in accordance with Guide 12, RCS STABILIZATION

C. (1) Isolate both OTSGs
(2) Go to OP-TM-EOP-009, HPI COOLING

D. (1) Isolate both OTSGs
(2) Cycle the PORV to maintain SCM in accordance with OP-TM-EOP-005

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The 'B' OTSG tube leak is worse than the 'A' OTSG based on radiation monitor readings and OTSG levels; (2) When offsite dose approaches 500 mRem TEDE (OP-TM-EOP-005, Rev 9, page 17), the operating crew must isolate the most affected OTSG when RCS pressure is less than 1000 psig; (3) Due to the LOOP, all Circulating Water Pumps trip which will cause a Condenser Vacuum Interlock (CVI). When CVI actuates, the Turbine Bypass Valves (MS-V-3A-F) latch closed. (4) With the MS-V-3's closed, the only way to control OTSG pressure is with the Atmospheric Dump Valves (MS-V-4's) on the OTSG which is NOT isolated.		
A.	(1) Isolate OTSG 'B', only (2) Throttle open MS-V-4A, OTSG 'A' Atmospheric Dump Valve, in accordance with Guide 12, RCS STABILIZATION	CORRECT ANSWER: The examinee will have to diagnose that the 'B' OTSG has a worse tube leak and must be the OTSG that is isolated. In addition, the examinee must know that MS-V-4A is the only way to control steam pressure on the 'A' OTSG due to the Loss of Offsite Power.
B.	(1) Isolate OTSG 'B', only (2) Throttle open MS-V-3D/E/F, OTSG 'B' Turbine Bypass Valves, in accordance with Guide 12, RCS STABILIZATION	INCORRECT: (1) Correct Answer. (2) Incorrect because the LOOP actuated CVI. Plausible if the examinee does not understand that on a loss of offsite power that the condenser is unavailable to steam to.
C.	(1) Isolate both OTSGs (2) Go to OP-TM-EOP-009, HPI COOLING.	INCORRECT: (1) Incorrect because the procedure step 3.35 only directs isolation of the most affected OTSG. Plausible if the examinee believes both OTSG's can be isolated with RCS pressure is < 1000 psig. (2) Incorrect because both OTSGs are not being isolated. Both OTSGs would be isolated if both were challenging isolation criteria of >85% in the operating range. Plausible because Step 3.34 directs going to OP-TM-EOP-009 if both OTSGs are isolated.
D.	(1) Isolate both OTSGs (2) Cycle the PORV to maintain SCM in accordance with OP-TM-EOP-005	CORRECT: (1) Incorrect because the procedure step 3.35 only directs isolation of the most affected OTSG. Plausible if the examinee believes both OTSG's can be isolated with RCS pressure is < 1000 psig. (2) Incorrect there is no need to cycle the PORV because pressure is already below 100 psig. Plausible if the examinee believes lowering RCS pressure is the man priority at this point.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039		A2.03
	Importance Rating	3.7		

K/A: Main and Reheat Steam: Ability to (a) predict the impacts of the following on the MRSS; and (b) based on the predictions, use procedures to correct, control, or mitigate the operations: Indications and alarms for main steam and area radiation monitors (during SGTR).

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Proposed Question:	Question #89		
Technical Reference(s):	OP-TM-EOP-005, Rev 9		
Proposed References to be provided to applicants during examination:			None
Learning Objective:	EOP005 PCO4		
Question Source:	Bank #	1110826	
	Modified Bank #		
	New		
Question History:	18-01 Comp 3	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments:

KA Match: The question matches the KA because the examinee must predict the impact of the isolation of the 'B' OTSG will have on the plant. The examinee has to have knowledge of a decision point in OP-TM-EOP-005 to assess the radiation monitor readings and decide with OTSG must be isolated.

High Cog: The question is High Cog because the examinee must know the isolation criteria for an OTSG and that the condenser is not available to steam too, in addition to assessing which OTSG must be isolated.

SRO Only: This is an SRO only question because the Senior Operator examinee must have knowledge of a diagnostic step in OP-TM-EOP-005 involves a transition to an event-specific sub procedure (Attachment 1 of EOP-005). The examinee must identify that the 'B' OTSG has a worse tube leak than the 'A' OTSG and then assess how to isolate the 'B' OTSG with a LOOP.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

90 **ID: 2106629** **Points: 1.00**

Plant Conditions:

- The reactor is at 8% power and holding following isolation of an Instrument Air (IA) leak.
- The section of the IA header that feeds AH-E-29A, Diesel Generator Room "A" Fan Damper, has been isolated and completely depressurized to facilitate repair of a tubing failure.
- Operators have performed OP-TM-861-910, Emergency Ventilation of EG-Y-1A Room.

Given the above information, which one of the following identifies:
(1) The current operability status of EG-Y-1A, and
(2) The requirements with respect to Fire Door D107, EG-Y-1A Room Door?

A. (1) OPERABLE
(2) Compensatory actions for a blocked open Fire Door are required

B. (1) OPERABLE
(2) The Fire Door must remain closed and latched except for temporary passage

C. (1) INOPERABLE
(2) Compensatory actions for a blocked open Fire Door are required

D. (1) INOPERABLE
(2) The Fire Door must remain closed and latched except for temporary passage

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The dampers on AH-E-29A & B will fail closed on a loss of instrument air (OP-TM-AOP-028, LOSS OF INSTRUMENT AIR, Rev 9 Page 31). (2) The Emergency Diesel could still operate, but would be declared inoperable and Technical Specification 3.7.2 would be entered (OP-TM-AOP-0281, LOSS OF INSTRUMENT AIR BASIS DOCUMENT, Rev 7 Page 4). (3) In accordance with OP-TM-861-910, EMERGENCY VENTILATION OF EG-Y-1A ROOM (Rev 2 Page 2) the crew must notify the shift management to implement FSI's and applicable compensatory action due to fire doors being blocked open.

A.	(1) OPERABLE (2) Compensatory actions for a blocked open Fire Door are required	INCORRECT: (1) Plausible because compensatory actions are being taken to address the Emergency Diesel Ventilation. Incorrect because the Emergency Diesel is still inoperable. (2) Correct answer.
B.	(1) OPERABLE (2) The Fire Door must remain closed and latched except for temporary passage	INCORRECT: (1) Plausible if the examinee does not know that AH-E-29A is required for Emergency Diesel Operability. (2) Plausible because fire doors are normally operated like this. Incorrect because OP-TM-861-910 allows for them to be open with FSI's.
C.	(1) INOPERABLE (2) Compensatory actions for a blocked open Fire Door are required	CORRECT: See above.
D.	(1) INOPERABLE (2) The Fire Door must remain closed and latched except for temporary passage	INCORRECT: (1) Correct answer. (2) Plausible because fire doors are normally operated like this. Incorrect because OP-TM-861-910 allows for them to be open with FSI's.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	078		2.2.44
	Importance Rating			4.6

K/A: Instrument Air: Ability to determine operability and/or availability of safety related equipment.

Proposed Question:	Question #90	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	OP-TM-AOP-0281, Rev 7
	OP-TM-861-910, Rev 2	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	861-GLO-14	

Question Source:	Bank #	897077	
	Modified Bank #		

	New			
Question History:	Simulator Exam 6	Last NRC Exam:	12-01	

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

10 CFR Part 55 Content:	55.41	
	55.43	b.1

Comments:

KA Match: This question matches the KA because the examinee must be able to determine the operability of an Emergency Diesel Generator when its associated instrument air is lost.

High Cog: This question is high cog because the examinee must determine that the loss of instrument air to AH-E-29A makes the damper fail closed. The examinee must use that knowledge to determine the Emergency Diesel Generator is inoperable.

SRO only: This question is SRO only because the examinee must know the administration of the fire protection program, specifically compensatory actions for a fire door that must be open.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

91 **ID: 2096110** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Pressurizer Level LT-1 is selected as the controlling channel

EVENT:

- A leak develops on the reference leg of LT-1 causing pressurizer level to change 60 inches over 10 seconds

Following the event, assuming no operator action, MU-V-17, Pressurizer Level Control Valve will __(1)__ , and __(2)__ will be entered to mitigate the malfunction?

A. (1) open
(2) OP-TM-MAP-G0205, PZR LEVEL HI/LO

B. (1) close
(2) OP-TM-MAP-G0205, PZR LEVEL HI/LO

C. (1) open
(2) OP-TM-MAP-H0302, SASS MISMATCH

D. (1) close
(2) OP-TM-MAP-H0302, SASS MISMATCH

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The pressurizer uses a wet reference leg, which means a 0 DP when the pressurizer is full; (2) Normal Level setpoint is 220 inches; (3) As the reference leg loses water, the DP would lower which would be seen as a pressurizer level rise on the PZR Level indication; (4) Since indicated pressurizer level is rising, MU-V-17, Pressurizer Level Control valve, will close to maintain level at setpoint of 220 inches; (5) Actual pressurizer would lower due to letdown flow remaining constant and makeup flow lowering due to MU-V-17 closing; (6) The pressurizer level instruments do not SASS on any failure. (6) OP-TM-MAP-G0205 PZR LEVEL HI/LO (Rev 3 Page 1) directs the crew to place MU-V-17 in hand and control Pressurizer level. The alarm response also directs the crew to select a valid level signal.

A.	(1) open (2) OP-TM-MAP-G0205, PZR LEVEL HI/LO	INCORRECT: (1) Plausible if the examinee determines that indicated Pressurizer level will be lower. Incorrect because indicated pressurizer level will be higher and MU-V-17 will close. (2) Correct answer.
B.	(1) close (2) OP-TM-MAP-G0205, PZR LEVEL HI/LO	CORRECT: (1) Correct answer. (2) Correct answer.
C.	(1) open (2) OP-TM-MAP-H0302, SASS MISMATCH	INCORRECT: (1) Correct answer. (2) Plausible because this procedure would be used to select a good instrument if a SASS actuation occurred. Incorrect because the Pressurizer Level instruments do not SASS.
D.	(1) close (2) OP-TM-MAP-H0302, SASS MISMATCH to hand. maintain a new level.	INCORRECT: (1) Since indicated level is rising, MU-V-17 will close. This is plausible if the examinee does not know which way indicated level will go. (2) Plausible because this procedure would be used to select a good instrument if a SASS actuation occurred. Incorrect because the Pressurizer Level instruments do not SASS.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	011	A2.03
Importance Rating		3.8

K/A: Pressurizer Level Control: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of PZR level

Proposed Question: Question #91

Technical Reference(s): OP-TM-MAP-G0205, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-5

EXAMINATION ANSWER KEY

Question Source:	Bank #		
	Modified Bank # 862259		
	New		
Question History:	Unmod on System Exam 11	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:
KA Match: The K/A is matched because the examinee must demonstrate knowledge of the operational implications of PZR reference leg leak abnormalities as they apply to Pressurizer Level Control Malfunctions. In addition, the examinee must know the procedure which will be used to mitigate the malfunction.

High Cog: The question is at the Comprehension/Analysis cognitive level because the examinee must demonstrate understanding of how the Pressurizer Level control system detects level, and then determine how a failure of a reference leg effects actual pressurizer level to correctly answer the question. In addition, the examinee must take into account how an erroneous pressurizer level effects the pressurizer makeup valve which would close and cause actual level to lower.

SRO Only: The question requires the examinee to assess the plant conditions and know what procedure which must be used to mitigate the malfunction. The procedures are not major EOPs or AOPs

Plant conditions:

- 100% power.
- Pressurizer Level LT-1 is selected as the controlling channel.

Event:

- A leak develops on the reference leg of LT-1.

Following the event, the trend of indicated (LT-1) Pressurizer level will ___(1)___ and actual Pressurizer level will ___(2)___.

A. (1) rise
(2) lower

B. (1) rise
(2) remain the same

C. (1) lower
(2) rise

D. (1) lower
(2) remain the same

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

92 **ID: 2105983** **Points: 1.00**

Plant Conditions:

- Reactor power is being lowered from 100% due to a steam leak in the reactor building
- The reactor is tripped when RB pressure exceeded 2 psig

Event:

- Loss of Offsite Power (LOOP)

Conditions one minute after the Loss of Offsite Power:

- 1D and 1E 4160V busses are being powered their Emergency Diesel Generators
- MAP H-1-8 alarm ICS/NNI POWER LOST is illuminated
  - ICS AUTO POWER 30 amp fuse is blown
- Incore temperatures are 560F and rising with only EF-P-1 available
- MAP C0101, RADIATION LEVEL HI is illuminated
  - RM-A-5 HI, Condenser Vacuum Pump Radiation Monitor is in ALARM
- Pressurizer Level is lowing at 1 inch per minute

In accordance with OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which of the following procedures has the highest priority based on current conditions?

A. OP-TM-AOP-020, LOSS OF STATION POWER

B. OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER

C. OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER

D. OP-TM-EOP-005, OTSG TUBE LEAKAGE

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OS-24 (Rev 31 Page 5), OTSG Tube leakage can be confirmed if the plant is post trip and a valid unexpected alarm from offgas or the steam line radiation monitors come in. (2) Prioritization of EOP actions is in the order that follows: Station Blackout, Loss of Subcooling Margin, Excessive Primary-to-Secondary Heat Transfer, Lack of Primary-to-Secondary Heat Transfer.		
A.	OP-TM-AOP-020, LOSS OF STATION POWER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions.
B.	OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions.
C.	OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions. The crew must wait for natural circulation to build in prior to meeting the entry criteria for OP-TM-EOP-004.
D.	OP-TM-EOP-005, OTSG TUBE LEAKAGE	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	055	2.4.8
	Importance Rating		4.5

K/A: Condenser Air Removal: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question:	Question 92	
Technical Reference(s):	OS-24, Rev 31	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP005-PCO-2	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	

	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	b.5	

Comments:

KA Match: This question matches the KA because the examinee must know how AOP's and EOP's are used in conjunction with each other. The examinee must make the determination that the EOP should be entered from the radiation monitor on the that monitors the Condenser Air Removal System.

High Cog: This question is high cog because the examinee must assess the plant conditions and then choose the correct procedure to enter.

SRO only: This question is SRO only because the examinee must assess the plant conditions and choose the highest priority (hierarchy) procedure to enter.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

93 **ID: 2096148** **Points: 1.00**

Plant conditions:

- An approved release is in progress from Waste Gas Decay Tank (WDG-T-1A).

Event:

- All Power is lost to RM-A-7, Gaseous Waste Discharge Tank Monitor.

Which one of the following describes the correct action and the subsequent release of WDG-T-1A?

(Assume all compensatory actions are complete in accordance with the ODCM table 2.1-2)

RM-A-8 - Aux and Fuel Handling Building Exhaust Area Radiation Monitor
WDG-V-47 - Waste Gas Release Stop and Control Valve

A. WDG-V-47 remains open. Manually terminate the release in accordance with 1104-27, "WASTE DISPOSAL – GASEOUS" until RM-A-7 is restored to operable.

B. WDG-V-47 remains open. Continue the release in accordance with 1104-27, "WASTE DISPOSAL – GASEOUS", as long as RM-A-8 is operable.

C. Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "RAD MON SYSTEM TROUBLE". Release may resume as long as RM-A-8 is operable.

D. Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "RAD MON SYSTEM TROUBLE". RM-A-7 must be restored to operable before resuming the release.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with the MAIN ANNUNCIATOR PANEL C (MAP C, Rev 102A, C-3-1 alarm response, Rev 17) will alarm due to a loss of voltage to RM-A-7; (2) When that occurs, the crew must ensure tech spec and/or ODCM requirements are met while the rad monitor is out of service and ensure radioactive releases to the environment are terminated and not resumed until appropriate monitors are operable; (3) In accordance with CY-TM-170-300, OFFSITE DOSE CALCULATION MANUAL (ODCM) (Rev 5, Pages 24 and 27), operability is not required when discharges are positively controlled through the closure of WDG-V-47 or where RM-A-8, AH-FT-149, and AH-FT-150 are operable and RM-A-8 is capable of automatically closing WDG-V-47.		
A.	WDG-V-47 remains open. Manually terminate the release in accordance with 1104-27, "WASTE DISPOSAL – GASEOUS" until RM-A-7 is restored to operable.	INCORRECT: Plausible because the examinee could believe that only RM-A-8 (Aux and Fuel Handling Building Gas Are Rad Monitor) closes WDG-V-47. In addition, the examinee could believe when power is lost to RM-A-7 that the interlock will not actuate. Incorrect because when RM-A-7 loses power, WDG-V-47 will go closed.
B.	WDG-V-47 remains open. Continue the release in accordance with 1104-27, "WASTE DISPOSAL – GASEOUS", as long as RM-A-8 is operable.	INCORRECT: Plausible because the examinee could believe that only RM-A-8 (Aux and Fuel Handling Building Gas Are Rad Monitor) closes WDG-V-47. In addition, the examinee could believe when power is lost to RM-A-7 that the interlock will not actuate. Incorrect because when RM-A-7 loses power, WDG-V-47 will go closed.
C.	Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "RAD MON SYSTEM TROUBLE". Release may resume as long as RM-A-8 is operable.	CORRECT: See above.
D.	Ensure WDG-V-47 closes to isolate the release. Place RM-A-7 in DEFEAT in accordance with MAP C-3-1, "RAD MON SYSTEM TROUBLE". RM-A-7 must be restored to operable before resuming the release.	INCORRECT: Plausible because WDG-V-47 does isolate the release. In addition the examinee could believe that RM-A-7 is required for a waste gas release. Incorrect because as long as other ODCM requirements are met, the release can continue as long as RM-A-8 is operable.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	K/A #	071		A2.02
	Importance Rating			2.9

K/A: Waste Gas Disposal: Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (2) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Use of waste gas release monitors, radiation, gas flow and totalizer.

Proposed Question:	Question #91

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Technical Reference(s):	MAP C, Rev 102A		CY-TM-170-300, Rev 5	
Proposed References to be provided to applicants during examination:			None	
Learning Objective:	231-GLO-8			
Question Source:	Bank #	537973		
	Modified Bank #			
	New			
Question History:	Comp 1	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41			
	55.43	b.1		

Comments:

KA Match: This question matches the KA because the examinee must be able to predict the status of the waste gas release when power to RM-A-7 is released, and then know procedures which will be used to correct the malfunction. The malfunction is the loss of a waste gas release radiation monitor.

SRO only: The question is SRO only because the examinee must the condition of the facility license (ODCM) which govern the required equipment for a gaseous waste release.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

94 **ID: 2087602** **Points: 1.00**

Plant Conditions:

- Reactor is in a refueling shutdown condition
- All fuel is in the spent fuel pool

Sequence of Events:

- Two hours into the shift, a licensed Control Room Operator (CRO) slips, and becomes incapacitated
- The incapacitated CRO is 1 of 3 licensed CROs assigned the shift
- The following personnel help transport the CRO to the hospital:
  - 1 of the 2 (not including the Shift Manager) available Senior Reactor Operators (SRO) assigned to the shift (both qualified STA)
  - 1 of the 4 available Auxiliary Operators (AO) assigned to the shift
  - Maintenance personnel have assumed the duties of Fire Brigade Team Leader
  - Other than the Shift Manager, no other operations personnel are on site

In accordance with OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS, and assuming all watchstanders are fully qualified, which of the following is the action to take, if any, to ensure required minimum unit staffing?

A. Another AO must be called in to arrive within two hours.

B. Another RO must be called in to arrive within two hours.

C. Another SRO must be called in to arrive within two hours.

D. No action is required because minimum staffing levels are still met for all positions.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) The reactor is in a refueling shutdown conditions, which in accordance with Technical Specification 1.2.6 (Page 1-1, AMD 278) which means reactor coolant temperature is no more than 140F; (2) According to Technical Specification table 6.2-1 (Page 6-2, AMD 219), the minimum shift crew composition is: 1 SRO, 1 RO, and 1 Non-Licensed Auxiliary Operator; (3) But due to other regulatory obligations, site procedure OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS (Rev 13, Page 2) four auxiliary operators are required to be assigned to shift.		
A.	Another AO must be called in to arrive within two hours.	CORRECT: See above.
B.	Another RO must be called in to arrive within two hours.	INCORRECT: Plausible because above 200F three CRO's are required. Incorrect because below 200F two CRO's are required.
C.	Another SRO must be called in to arrive within two hours.	INCORRECT: Plausible because a normal shift consists of 2 SRO's and a Shift Manager. Incorrect because only 1 SRO is required less than 200F
D.	No action is required because minimum staffing levels are still met for all positions.	INCORRECT: Plausible because the examinee could believe that 4 AO's are not required when less than 200F. All other staffing requirements lower when less than 200F, but 4 AO's are required in both conditions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.1.5
	Importance Rating		3.9

K/A: Ability to use procedures relate to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question:	Question 94	
Technical Reference(s):	T.S Table 6.2-1, AMD 217	
	OP-TM-112-101-1002, Rev 13	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	Prewatch DBIG APCO-1	
Question Source:	Bank #	
	Modified Bank #	1147373
	New	

Question History:	N/A	Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	b.1	

Comments:

KA Match: This question matches the KA because the examinee must know requirements in the operations procedure to shift staffing.

High Cog: This question is high cog because the examinee must identify that in a refueling shutdown that RCS temperature is below 200F then apply the requirements of the shift staffing procedure to ensure proper personnel are on site.

SRO-only: This question is SRO only because they are responsible to ensure proper shift staffing levels at all times.

Plant Conditions:

- The reactor is defueled during a refueling outage.
- No fuel handling is in progress.

Sequence of Events:

- Two hours into the shift, an Auxiliary Operator slips, hits her head, and becomes unconscious while travelling through the Reactor Building.
  - She is 1 of 3 Auxiliary Operators (AO) on the shift.
- She is contaminated and is escorted to the hospital by the following personnel:
  - 1 of the 2 available Rad Pro Technicians on shift.
  - 1 of the 3 available Reactor Operators (RO) assigned to the shift.

Given the above information, which of the following is the action to take, if any, to ensure **Technical Specification** required minimum unit staffing?

A. Another AO should be called in to arrive within two hours.

B. Another RO should be called in to arrive within two hours.

C. Another RP technician should be called in to arrive within two hours.

D. No action is required because minimum staffing levels are still met for all positions.

Answer D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

95 **ID: 2106085** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Secondary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) To minimize load on the Auxiliary Transformers

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-EOP-020, COOLDOWN FROM OUTSIDE OF CONTROL ROOM is entered when there is a fire in the control room or relay room that has a potential to cause damage to safe shutdown required equipment; (2) Once the determination is made that the control room must be evacuated, this starts a sequence in which actions are taken to prevent or terminate undesirable actions which may be caused by the fire; (3) With the announcement at the end of the IMA's, control room operators and non-licensed operators (NLOs) are directed to perform actions within attachments to mitigate possible effects of the fire; (4) The NLOs respond in accordance with OP-TM-EOP-020, and OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which allows to them to take actions prior to having CRS concurrence (Attachment F, of OS-24), which includes performing Attachment 5 and Attachment 13 of OP-TM-EOP-020; (5) The Secondary Safe Shutdown NLO will perform Attachment 13, which trips the reactor coolant pumps at the 7kV busses to prevent them from spuriously starting without seal cooling.

A.	(1) Attachment 5, "Preventing Spurious Operation of MOV's" (2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.	Incorrect Answer - Plausible because the MS-V-8's (Isolation valves to the Turbine Bypass Valves) are closed to prevent an overcooling event due the turbine bypass valves failing midscale on some fire events. In addition the correct reason for performing Attachment 5 is to terminate uncontrolled HPI.
B.	(1) Attachment 5, "Preventing Spurious Operation of MOV's" (2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.	Incorrect Answer - Plausible because the MS-V-8's (Isolation valves to the Turbine Bypass Valves) are closed to prevent an overcooling event due the turbine bypass valves failing midscale on some fire events. The MS-V-2's could also be used to isolate the turbine bypass valves, but closing the MS-V-2 would also isolate a steam supply path to EF-P-1 (Steam Driven Emergency Feedwater Pump) and an MS-V-4 (Atmospheric Dump Valve). Attachment 5 does open the breaker for MS-V-2A/B, but that is to maintain the valve open for use of EF-P-1 and the MS-V-4's.
C.	(1) Attachment 13, "Tripping RCPs Locally" (2) To minimize load on the Auxiliary Transformers	Correct Answer - Plausible because the reactor coolant pumps are a major load on the auxiliary transformers.
D.	(1) Attachment 13, "Tripping RCPs Locally" (2) Prevent Reactor Coolant pumps from operating with no seal injection.	Correct Answer - See above.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		1
K/A #	2.1.8	
Importance Rating		4.1

K/A: Ability to coordinate personnel activities outside the control room.

EXAMINATION ANSWER KEY

Proposed Question: Question #95

Technical Reference(s): OP-TM-EOP-020, Rev 22 OS-24, Rev 31

OP-TM-EOP-0201, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-PCO-1

Question Source: Bank #

Modified Bank # 1720770

New

Question History: unmod on 18-01 Cert Exam Last NRC Exam: unmod on 16-01 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

KA Match: This question matches the KA because the examinee will have to know where various personnel are required to go when the control room must be evacuated.

High Cog: This question is high cog because the examinee will have to assess plant conditions to determine that a control room evacuation is required and that the remote shutdown sequence has been directed. Step 2.8 is a stopping point in which the operating crew would wait at until conditions in the control room have deteriorated enough that crew must evacuate. This is denoted by the tripping of the feedwater, condensate booster, and condensate pumps.

SRO Only: This question is SRO only because the examinee will have to assess that the stem in a way that determines that the remote shutdown sequence has begun and understand which attachment the NLOs have to perform.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Primary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent Reactor Coolant pumps from operating with no seal injection.

Answer: A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

96 **ID: 2088112** **Points: 1.00**

In accordance with AD-AA-101, PROCESSING OF PROCEDURES AND T&RMs a(n) __(1)__ is a non-permanent procedure change that does contain a change of intent.

The procedure is a site-specific procedure that does NOT deviate from a fleet standard procedure or T&RM requirements.

Approval from a __(2)__ is required.

A. (1) Interim Change
(2) Station Qualified Reviewer, only

B. (1) Interim Change
(2) Station Qualified Reviewer and Site Functional Area Manager

C. (1) Temporary Change
(2) Qualified SRO, only

D. (1) Temporary Change
(2) Qualified SRO and a Station Qualified Reviewer

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with AD-AA-101 (Rev 29) that an INTERIM CHANGE (Page 2) is a non-permanent document change that contains a change of intent. They must also know that TEMPORARY CHANGE (Page 4) is a non-permanent procedure change that does NOT contain a change of intent; (2) According to step 4.2.1 (page 15), if the procedure is an Interim Change then use form AD-AA-101-F-01, DOCUMENT SITE APPROVAL. AD-AA-101-F-01 requires an SQR (Rev 7 Page 1) and SFAM (Page 2) approval.

A.	(1) Interim Change (2) Station Qualified Reviewer, only	INCORRECT: (1) Correct answer. (2) Plausible because an SQR is required. Incorrect because the SFAM is required as well.
B.	(1) Interim Change (2) Station Qualified Reviewer and Site Functional Area Manager	CORRECT: See above.
C.	(1) Temporary Change (2) Qualified SRO, only	INCORRECT: (1) Plausible because there are only two types of changes, INTERIM and TEMPORARY. Incorrect because the revision contains a change of intent. It must be processed as an INTERIM CHANGE. (2) Plausible because an SROs must approve procedure changes. Incorrect because they are not the only personnel who approve them.
D.	(1) Temporary Change (2) Qualified SRO and a Station Qualified Reviewer	INCORRECT: (1) Plausible because there are only two types of changes, INTERIM and TEMPORARY. Incorrect because the revision contains a change of intent. It must be processed as an INTERIM CHANGE. (2) Plausible because a Qualified SRO and Station Qualified Reviewer would be required to process a Temporary Change.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.2.6
	Importance Rating		3.6

K/A: 2.2.6 - Knowledge of the process for making changes to procedures.

Proposed Question:	Question #96	
Technical Reference(s):	AD-AA-101, Rev 29	
	AD-AA-101-F-10	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EQC00016	
Question Source:	Bank #	
	Modified Bank #	

	New	X			
Question History:	N/A		Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41				
	55.43	b.3			

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the permissions required to process a temporary procedure change.

SRO only: This question is an SRO only question because it involves knowing the process for changing plant procedures. This is an SRO task.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

97 **ID: 2105938** **Points: 1.00**

In accordance with WC-AA-101-1004, ON-LINE MAINTENANCE FOR LIMITING CONDITION FOR OPERATION OF SYSTEMS OR COMPONENTS, it is required to continuously work critical path for LCOs of _______. (Maximum LCO time this rule is in effect)

A. 24 hours

B. 48 hours

C. 72 hours

D. 7 days

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with WC-AA-101, ON-LINE WORK CONTROL PROCESS (Rev 28, Page 17), Attachment 1, a 7 day LCO is required to be worked 24 hours a day.		
A.	24 hours	INCORRECT: Plausible because a 24 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies.
B.	48 hours	INCORRECT: Plausible because a 48 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies
C.	72 hours	INCORRECT: Plausible because a 72 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies
D.	7 days	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			3
	K/A #			2.2.19
	Importance Rating			3.4

K/A: Knowledge of maintenance work order requirements

Proposed Question:	Question #97	
Technical Reference(s):	WC-AA-101, Rev 28	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	NOP DPBIG-APCO-3	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	b.1

Comments:

KA Match: This question matches the KA because the examinee must know when maintenance must be continuously worked, which would be a maintenance work order requirement.

SRO only: This question is SRO only because the examinee must know that if the work being performed would cause the Unit to enter an LCO of 7 days or less that the planned work must be continuously worked.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

98 **ID: 2105798** **Points: 1.00**

Plant Conditions:

- Reactor power is 25% with ICS in auto

Event:

- Personnel must enter the 'A' D-ring to investigate RC-P-1A for an oil leak

Given the above information, which of the following describes the minimum personnel required to authorize this work to be performed?

A. Shift Manager

B. Shift Manager and Radiation Protection Manager

C. Shift Manager, Radiation Protection Manager, and Plant Manager

D. Shift Manager, Radiation Protection Manager, Plant Manager and Site Vice President

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with RP-TM-460-1007, ACCESS TO TMI-1 REACTOR BUILDING (Rev 8, Page 4) approval of the Shift Manager, Plant Manager, and Radiation Protection Manager are required for entries within the secondary shield.		
A.	Shift Manager	INCORRECT: Plausible because his permission is required. Incorrect because the Plant Manager and Radiation Protection Manager are also required.
B.	Shift Manager and Radiation Protection Manager	INCORRECT: Plausible because their permission is required. Incorrect because the Plant Manager is also required to give permission.
C.	Shift Manager, Radiation Protection Manager, and Plant Manager	CORRECT: See above.
D.	Shift Manager, Radiation Protection Manager, Plant Manager and Site Vice President	INCORRECT: Plausible because the Shift Manager, Radiation Protection Manager and Plant Manager are required. Incorrect because the Site Vice President is not required.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	2.3.12		
	Importance Rating	3.7		

K/A: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:	Question #98	
Technical Reference(s):	RP-TM-460-1007, Rev 8	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	NOP-DBIG-APCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge			

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	b.4	

Comments:

KA Match: This question matches the KA because the examinee must know have knowledge of the procedure which allows personnel into the reactor building at power. The examinee must know the containment entry requirements to allow work in the 'A' D-ring.

High Cog: This question is high cog because the examinee must assess the plan conditions and recall whose permission is required for work in the 'A' D-ring.

SRO Only: This question is SRO only because the examinee must have knowledge of the radiation hazard posed by work on the polar crane at power. The examinee must know a special level of authorization is required to perform work in this area.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

99 **ID: 2105849** **Points: 1.00**

In accordance with EP-AA-112-F-01, COMMAND AND CONTROL TURNOVER BRIEFING FORM, which of the following duties CANNOT be delegated to the EOF?

A. NRC Notifications

B. Event Classification

C. PAR Decision-Making

D. State/Local Notifications

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

Explanation: To answer this question correctly, the examinee must know: In accordance with EP-AA-112-F-01 (Rev G Page 1), Event Classification is not a delagatable event to the EOF.		
A.	NRC Notifications	INCORRECT: Plausible because this can be delegated to the TSC and EOF.
B.	Event Classification	CORRECT: See above.
C.	PAR Decision-Making	INCORRECT: Plausible because this can be delegated to the TSC and EOF.
D.	State/Local Notifications	INCORRECT: Plausible because this can be delegated to the TSC and EOF.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.37	
	Importance Rating	4.1	

K/A: Knowledge of the lines of authority during implementation of the emergency plan

Proposed Question:	Question #99	
Technical Reference(s):	EP-AA-112-F-01, Rev G	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP-DBIG-PCO-6	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee must know the lines of authority of the command and control checklist during an Eplan event.

SRO Only: This question is SRO only because the examinee must have knowledge of organizational hierarchy during an Eplan event. The examinee must know that event classification can only be performed in the control room or TSC.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM SUBMITTAL - 3/27

100 **ID: 2095211** **Points: 1.00**

Plant Conditions:

- Refueling Operations in progress; Fuel is being removed from the core
- DH Loop B in operation

Event:

- DH-P-1B trips on overcurrent
- Incore and RCS temperatures have risen from 124°F to 132°F and are currently steady

Based on these conditions, identify the ONE selection below that describes:

(1) Whether a Reactor Operating Mode change has occurred.
(2) Procedure to be used to respond to the event.

A. (1) A Reactor Operating Mode change has occurred
(2) EOP-030, Loss of Decay Heat Removal

B. (1) A Reactor Operating Mode change has NOT occurred
(2) EOP-030, Loss of Decay Heat Removal

C. (1) A Reactor Operating Mode change has occurred
(2) OP-TM-212-111, SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATION

D. (1) A Reactor Operating Mode change has NOT occurred
(2) OP-TM-212-111, SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATION.

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When DH-P-1B trips, since the fuel is not fully unloaded, the RCS temperature could rise; (2) Initially, the crew would enter the alarm response OP-TM-MAP-B0102, 4KV ES MOTOR TRIP , which would direct the crew to initiate OP-TM-211-111, SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATION; (3) The mode change from REFUELING SHUTDOWN to COLD SHUTDOWN when RCS Temperature rises above 140°F, which does NOT happen in the stem of the question.		
A.	(1) A Reactor Operating Mode change has occurred. (2) EOP-030, Loss of Decay Heat Removal.	Incorrect Answer: (1) Plausible if the examinee believes a mode change occurred at 130F. Incorrect because it changes at 140F. (2) Plausible because the examinee could believe that EOP-030 must be entered with the DH-P-1B trips. Incorrect because the entry criteria for this is either both trains of DHR unavailable or a >10F temperature change, neither of which exist for the stem of this question.
B.	(1) A Reactor Operating Mode change has NOT occurred. (2) EOP-030, Loss of Decay Heat Removal.	Incorrect Answer: (1) Correct answer. (2) Plausible because the examinee could believe that EOP-030 must be entered with the DH-P-1B trips. Incorrect because the entry criteria for this is either both trains of DHR unavailable or a >10F temperature change, neither of which exist for the stem of this question.
C.	(1) A Reactor Operating Mode change has occurred. (2) OP-TM-212-111, SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATION	Incorrect Answer: (1) Plausible if the examinee believes a mode change occurred at 130F. Incorrect because it changes at 140F. (2) Correct Answer.
D.	(1) A Reactor Operating Mode change has NOT occurred. (2) OP-TM-212-111, SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATION.	Correct Answer: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.4.9	
	Importance Rating		4.2

K/A: Knowledge of lower power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: Question #100

Technical Reference(s): OP-TM-MAP-B0102, Rev 2 OP-TM-EOP-030, Rev 11

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-030-PCO-2

Question Source: Bank #

Modified Bank # 1740783

New

Question History: Last NRC Exam: X Unmodified on 16-01 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

KA Match: This question matches the KA because the examinee must know the mitigation strategy for a loss of decay heat removal during a shutdown situation. They must also be cognizant of operating mode, which could lead to changing shutdown cooling requirements.

High Cog: This question is high cog because the examinee must analyze conditions in the stem and choose the correct course of action.

SRO only: This question is SRO only because the examinee must know what the tech spec definitions for REFUELING SHUTDOWN and COLD SHUTDOWN, and they apply to the stem of the question.

Plant Conditions:

- Refueling Operations in progress.
- DH Loop A in operation.

Event:

- DH-P-1A trips on overcurrent.
- Incore and RCS temperatures have risen from 134°F to 145°F and are currently steady.

Based on these conditions, identify the ONE selection below that describes:

(1) Whether a Reactor Operating Mode change has occurred.
(2) Procedure to be used to respond to the event.

A. (1) A Reactor Operating Mode change has occurred.
(2) EOP-030, Loss of Decay Heat Removal.

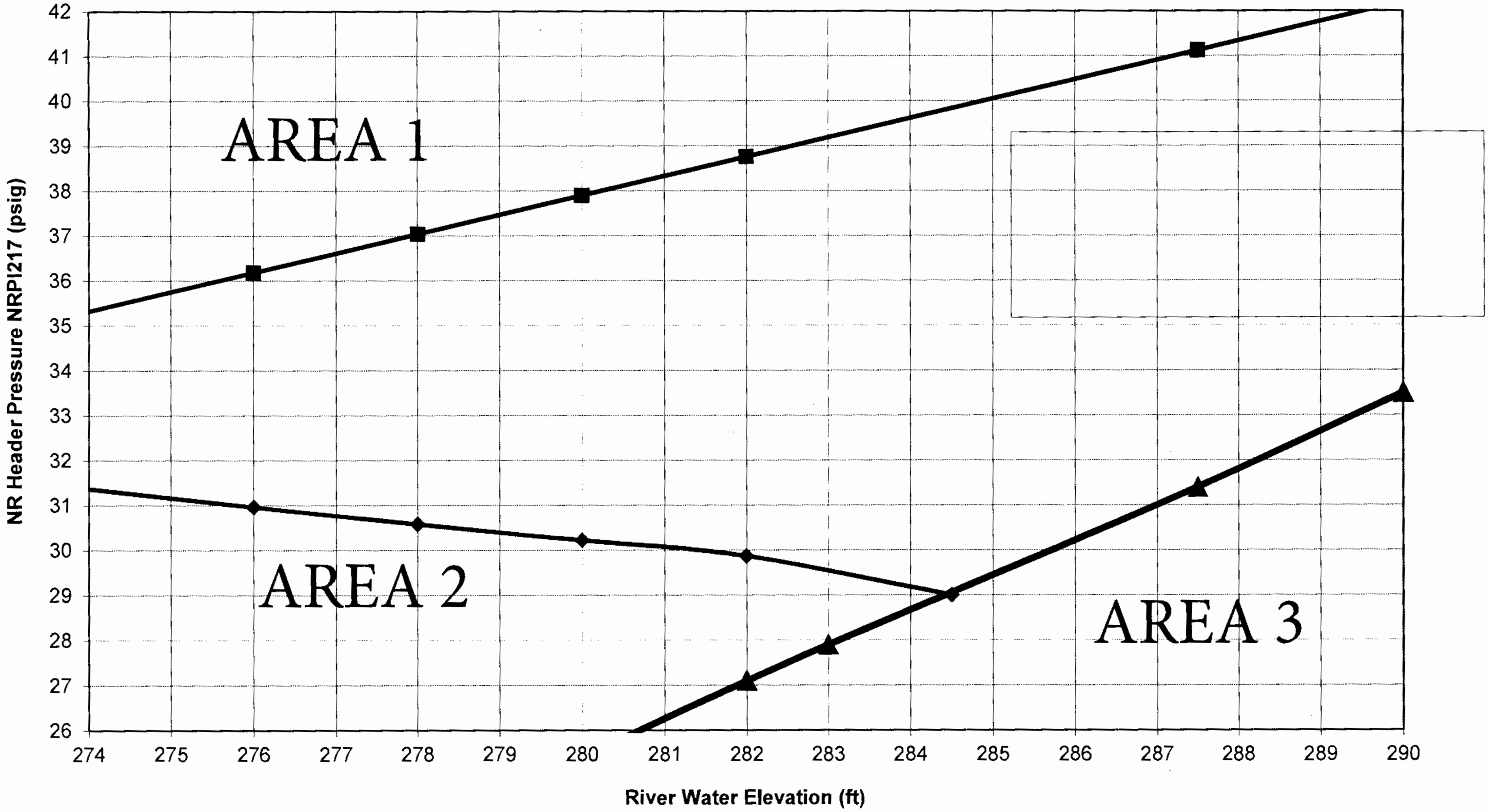
B. (1) A Reactor Operating Mode change has NOT occurred.
(2) EOP-030, Loss of Decay Heat Removal.

C. (1) A Reactor Operating Mode change has occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

D. (1) A Reactor Operating Mode change has NOT occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

Answer: A

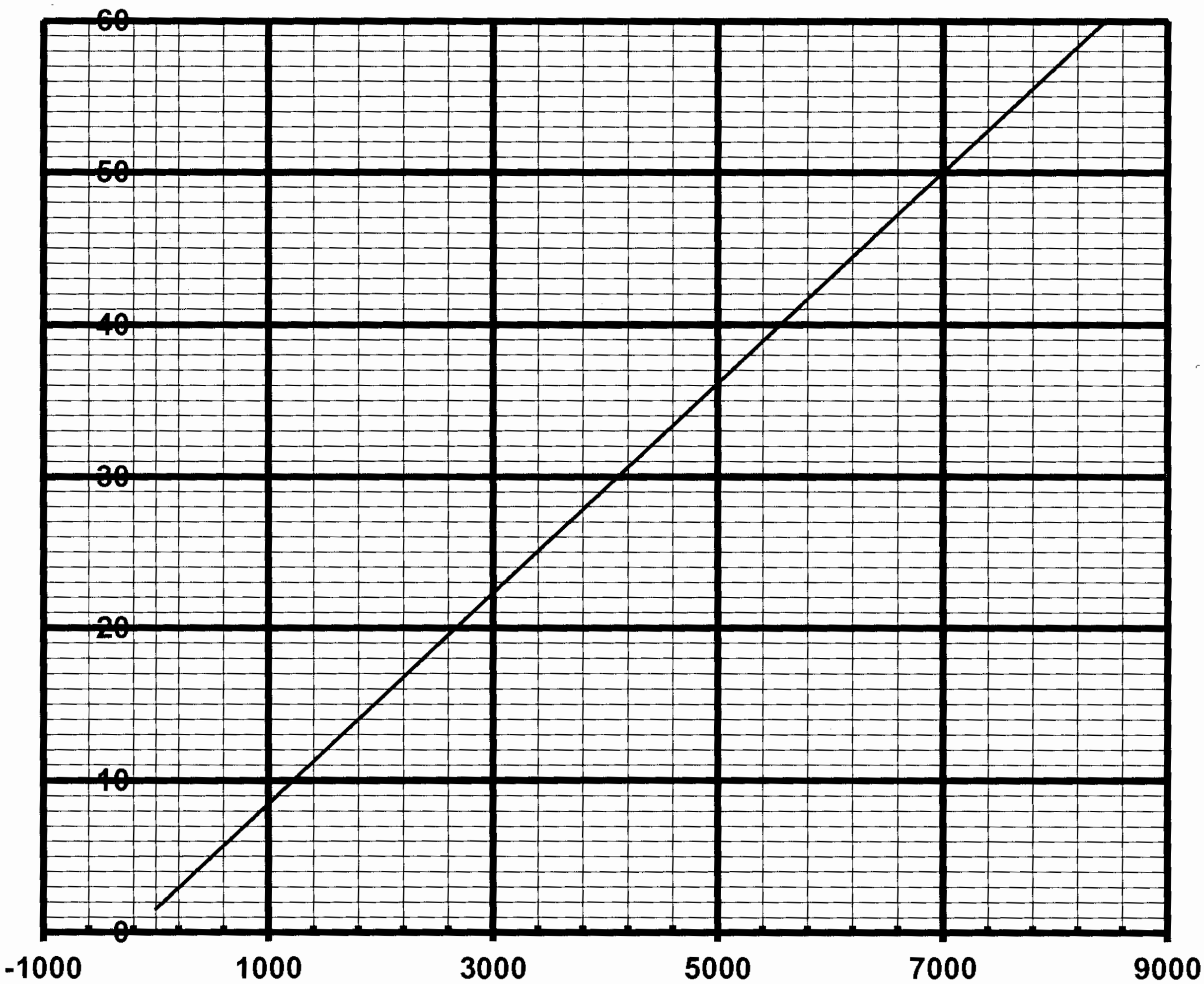
ATTACHMENT 7.2
NR System Operating Band
Page 1 of 1

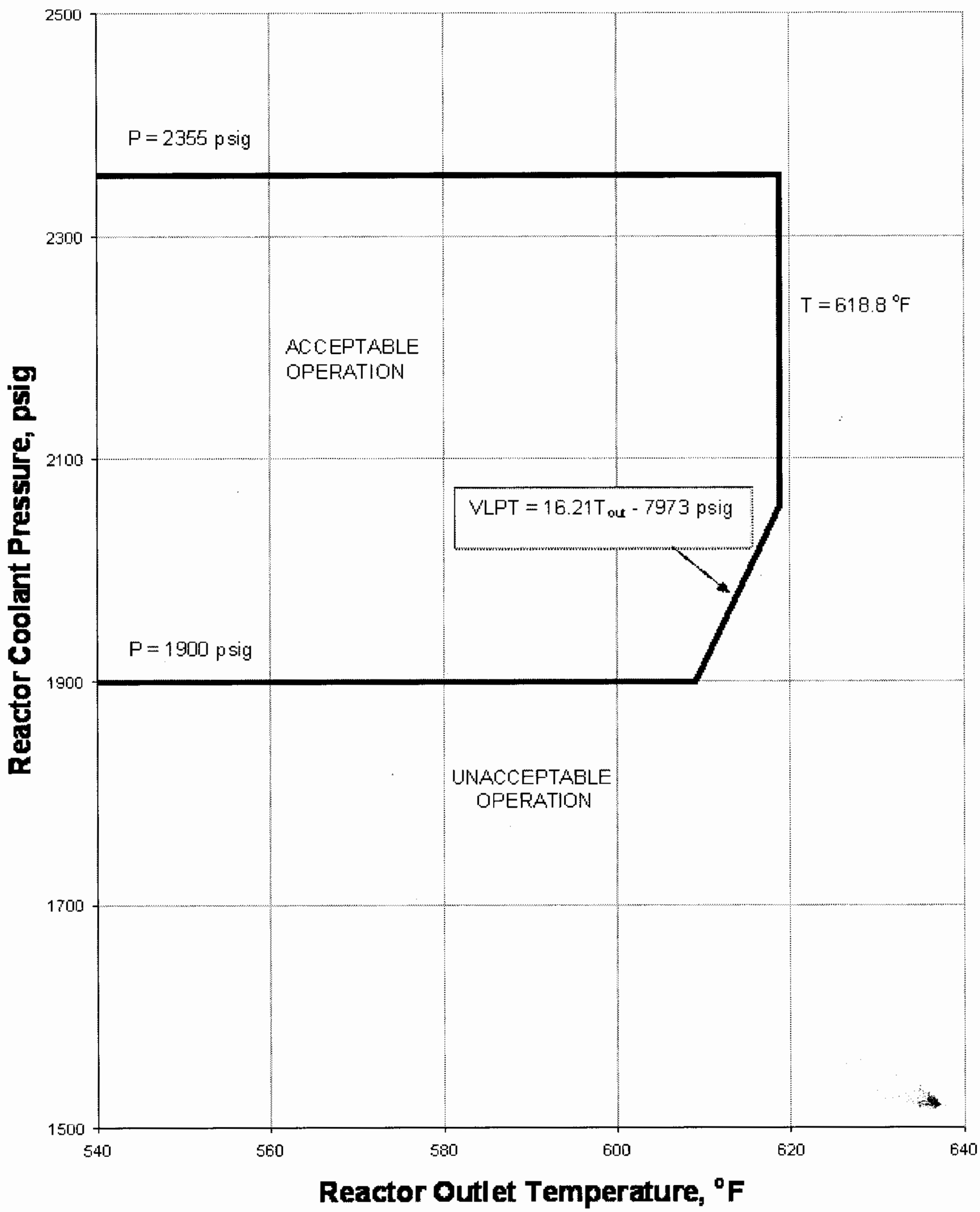


ATTACHMENT 7.2
Minimum Height of Water Required to Avoid Vortex Formation vs. Decay Heat Flow
Page 1 of 1

NOTE: The vortex formation curve includes instrument error (Ref. Memo 5810-94-0080) and is given by the following expressions:

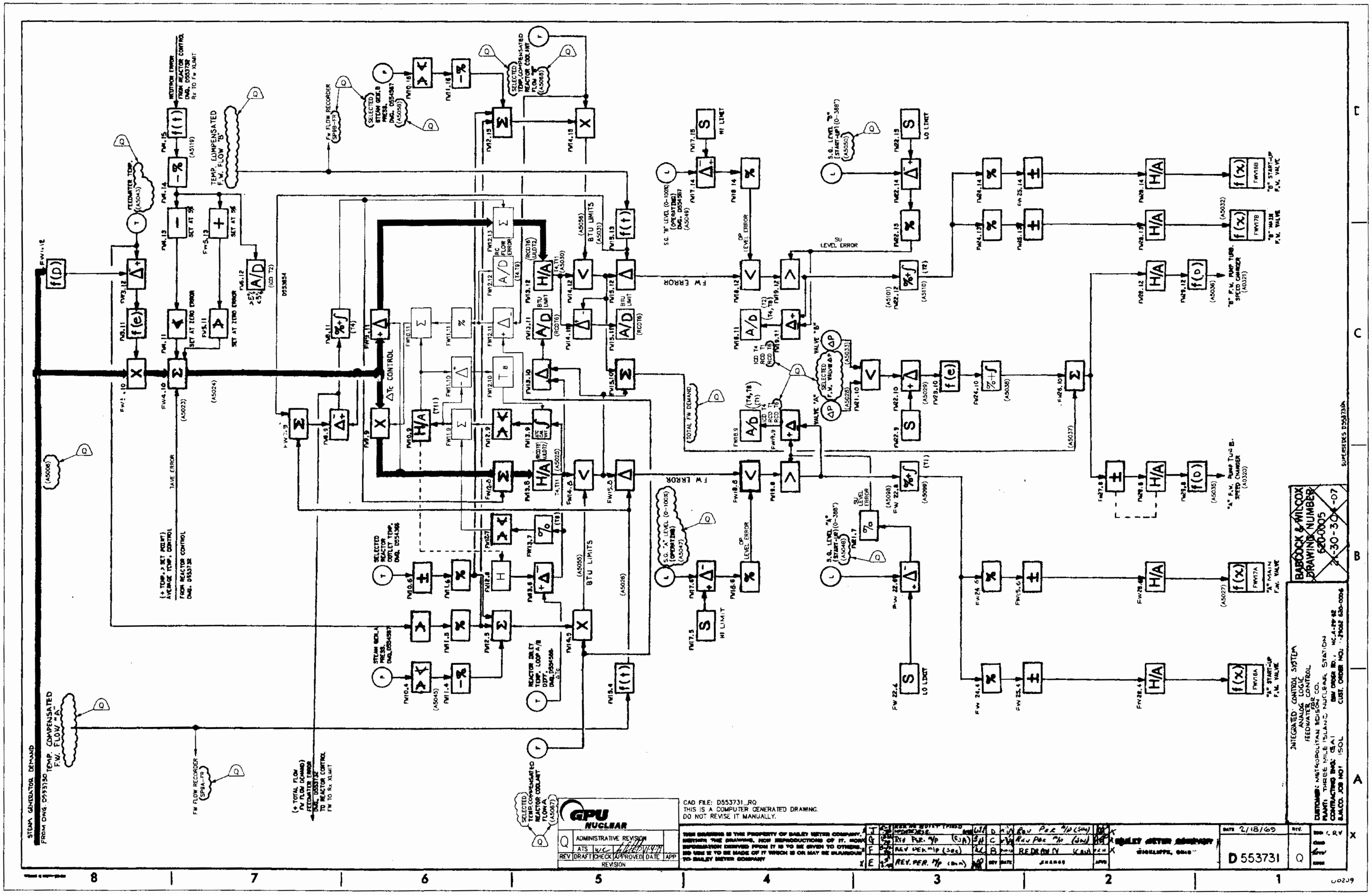
Minimum Draindown Level (inches) = 0.00692 (Total DH Flow in gpm) + 1.54
or
Maximum Total DH Flow (gpm) = 144.5 (Inches Draindown Level – 1.54)





PROTECTION SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
TMI-1
FIGURE 2.3-1

Amendment No. ~~13, 17, 28, 39, 45,~~
~~78, 126, 135, 142, 167, 247~~, 262



INTEGRATED CONTROL SYSTEM
ANALOG LOGIC
FEEDWATER CONTROL
CUSTOMER: METROPOLITAN EDISON CO.
PLANT: THREE MILE ISLAND NUCLEAR STATION
BABCOCK & WILCOX
DRAWING NUMBER
SUPERSEDES D553731A
FW ERROR
BTU LIMITS
TOTAL FW DEMAND
STEAM GENERATOR DEMAND
TEMP. COMPENSATED F.W. FLOW "A"
TEMP. COMPENSATED F.W. FLOW "B"
FW FLOW RECORDER
ΔTc CONTROL
LO LIMIT
HI LIMIT
SU LEVEL ERROR
OP LEVEL ERROR
CAD FILE: D553731_RQ
THIS IS A COMPUTER GENERATED DRAWING.
DO NOT REVISE IT MANUALLY.
GPU NUCLEAR
ADMINISTRATIVE REVISION
REV
DRAFT
CHECK
APPROVED
DATE
APP
REVISION
BAILEY METER COMPANY
DATE 2/18/69
D 553731
Q
8
7
6
5
4
3
2
1
D
C
B
A