KEY VERIFY RESCORE ⊏R⊐

EXAM KEY

TMI ILT 10-2

~~SRO~~ VERSION
4/25/12 RO

IMPORTANT

USE NO. 2 PENCIL ONLY

- EXAMPLE: ⊏A⊐ ⊏B⊐ ▬ ⊏D⊐ ⊏E⊐
- MAKE **DARK** MARKS
- ERASE **COMPLETELY** TO CHANGE
- MAKE NO STRAY MARKS

ID NUMBER

(T) (F)

1	B	51	B
2	B	52	D
3	B	53	C
4	D	54	C
5	B	55	B
6	C	56	C
7	A	57	B
8	A	58	B
9	A	59	C
10	D	60	D
11	A	61	A
12	D	62	C
13	C	63	C
14	B	64	C
15	D	65	C
16	B	66	B
17	B	67	C
18	C	68	C
19	B	69	A
20	B	70	C
21	C	71	A
22	C	72	D
23	C	73	D
24	B	74	A
25	B	75	C
26	D	76	
27	C	77	
28	D	78	
29	B	79	
30	A	80	
31	D	81	
32	A	82	
33	D	83	
34	B	84	
35	B	85	
36	A	86	
37	B	87	
38	A	88	
39	D	89	
40	C	90	
41	D	91	
42	D	92	
43	C	93	
44	C	94	
45	B	95	
46	A	96	
47	A	97	
48	D	98	
49	C	99	
50	D	100	

RESCORE

KEY 75

SCORE

KEY VERIFY RESCORE R

(T) A	(F) B	C	D	Marked

No.	Marked	No.	Marked
1	B	51	B
2	B	52	D
3	B	53	C
4	D	54	C
5	B	55	B
6	C	56	C
7	A	57	B
8	A	58	B
9	A	59	C
10	D	60	D
11	A	61	A
12	D	62	C
13	C	63	C
14	B	64	C
15	D	65	C
16	B	66	B
17	B	67	C
18	C	68	C
19	B	69	A
20	B	70	C
21	C	71	A
22	C	72	D
23	C	73	D
24	B	74	A
25	B	75	C
26	D	76	B
27	C	77	C
28	D	78	C
29	B	79	B
30	A	80	A
31	D	81	B
32	A	82	D
33	D	83	B
34	B	84	B
35	B	85	D
36	A	86	C
37	B	87	A
38	A	88	B
39	D	89	A
40	C	90	B
41	D	91	D
42	D	92	B
43	C	93	B
44	C	94	D
45	B	95	B
46	A	96	B
47	A	97	D
48	D	98	C
49	C	99	B
50	D	100	A

IMPORTANT

USE NO. 2 PENCIL ONLY

- EXAMPLE: A B ● D E
- MAKE ***DARK*** MARKS
- ERASE ***COMPLETELY*** TO CHANGE
- MAKE NO STRAY MARKS

ID NUMBER

0	0	0	0	0	0	0	0	0	0
1	1	1	1	1	1	1	1	1	1
2	2	2	2	2	2	2	2	2	2
3	3	3	3	3	3	3	3	3	3
4	4	4	4	4	4	4	4	4	4
5	5	5	5	5	5	5	5	5	5
6	6	6	6	6	6	6	6	6	6
7	7	7	7	7	7	7	7	7	7
8	8	8	8	8	8	8	8	8	8
9	9	9	9	9	9	9	9	9	9

EXAM KEY

TMI ILT 10-2
SRO VERSION

RESCORE

KEY 100

SCORE

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K1.01	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: 120V vital/instrument power system

Proposed Question: RO Question # 1

Plant conditions:

- Plant is at 70% power.
- All four (4) RC Pumps are running.
- VBA is de-energized due to failure of the 1A Inverter.
- One (1) hour is the time estimate for restoration of VBA.
- During investigation of the Inverter failure, RC-P-1C #1 seal fails.

Which ONE (1) of the following statements describes how the plant will respond if RC-P-1C is secured within 30 minutes while at 70% power?

A. ICS will run the plant back to 50% power and re-ratio Main Feedwater.

B. All operable RPS channels will trip on flux/pumps, causing a reactor trip.

C. The plant will remain at 70% power and ICS will re-ratio Main Feedwater.

D. RPS channel "C" will trip on flux/pumps when RC-P-1C is secured, and ICS will re-ratio Main Feedwater at 70% power.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-MAP-H0101 (p1; Rev 1), the ICS will runback the unit to 405 MWe (50% NI power) at a rate of 50%/minute on a loss of 2 RCPs.

B. **Correct.** According to TQ-TM-104-641-C001 (p77; Rev 1), on the loss of VBA the A RPS channel will trip. Additionally, the other 3 RPS channels will see the RC-P-1A as tripped, and CRD-CB-1 will trip. According to OS-24 ATTACHMENT A (p2; Rev 19), a reactor trip is required and will automatically occur if Reactor Power >55% with less

than 3 RCPs operating. According to OP-TM-AOP-040 (p3; Rev 0), if an RCP Seal Failure occurs, the operator will be directed to reduce power to less than 75% within 30 minutes to support RCP Shutdown. If RC-P-1C is shutdown at 70% power, the RPS will see power at >55% with < 3 RCPs running and generate a reactor trip.

C. **Incorrect.** This is plausible because according to TQ-TM-104-621-C001 (p147; Rev 2), upon loss of a Reactor Coolant Pump, a reduction in Reactor Coolant Flow (RC Flow) occurs within the affected loop but rises in the non-affected loop. Within ten seconds RC Flow in the affected loop has lowered to approximately 32×10^6 lbm/hr., while flow in the non-affected loop rises to approximately 76×10^6 lbm/hr. The differential flow developed between A and B loops will activate the DT_C control circuit, and swap T_{avg} control to the loop with the greatest flow. The DT_C control has a circuit sensitive to RC Flow Differential, if greater than 10%. This circuit offers rate sensitive/promotional control. Thus, an immediate response to re-ratio feedwater demand proportionate with the rate of change to RC Flow occurs.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that only the C RPS channel will be affected by the given events, not recognizing that according to TQ-TM-104-641-C001 (p73; Rev 1), each RCP power monitor feeds signals to all four RPS channels.

Technical Reference(s):	TQ-TM-104-641-C001 (p77; Rev 1) OS-24 ATTACHMENT A (p2; Rev 19) OP-TM-AOP-040 (p3; Rev 0) OP-TM-MAP-H0101 (p1; Rev 1) TQ-TM-104-621-C001 (p148; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-8 (As available)

Question Source:	Bank #	QR-641-GLO-8-Q03	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the physical connections and/or cause effect relationships between the RPS and the 120V vital/instrument power system (I.e.VBA).

The question is at the Comprehension/Analysis cognitive level because the operator must use knowledge of the RCP Monitors including their power supplies, and knowledge of the RPS channels (i.e. what is the logic/setpoints for flux/power reactor trip), and evaluate plant conditions, to determine the correct answer.

What MUST be known:
1. The power supplies to the RCP Monitors. 2. The logic of the RCP monitors into the RPS modules. 3. The setpoint of the automatic flux/power Reactor Trip. 4. How the loss of VBA will affect the RCP monitors and RPS modules.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K1.02	
	Importance Rating	3.4		

Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: MFW System

Proposed Question: RO Question # 2

Which ONE (1) of the following describes the operation of the Main Feedwater Pump Monitoring circuit that provides Emergency Feedwater an input from the Heat Sink Protection System?

EACH PUMP uses.....

A. two pressure bistables to sense Main Feedwater Pump discharge pressure.

B. two pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

C. three pressure bistables to sense Main Feedwater Pump discharge pressure.

D. three pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the presence of Feed Pump discharge pressure would be indicative of the operating status of the Main Feed Pump. The operator may incorrectly believe that the parameter sensed to determine that the FW Pump is operating or not is the discharge pressure.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-644-C001 (p13; Rev 1), the operating status of the Main Feedwater Pumps is monitored by nine pressure bistables, per pump, two of which are associated with the HSPS. The bistables sense hydraulic oil pressure at the feedwater pump turbine stop valves. Normal pressure is approximately 200 psig, and an oil pressure of < 75 psig will signal the circuit that the feed pump has tripped. If both bistables in either Train A or Train B

close, one train of EFW pumps starts and one EF-V-30 per OTSG will control at 25 inches on Startup Range level. If all four bistables close, all EFW pumps start and two EF-V-30's per OTSG will control at 25 inches on Startup Range level.

C. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-644-C001 (p12; Rev 1), of the nine bistables per pump, three of them are associated with the Main Turbine Trip Circuit. The operator may incorrectly believe that the three bistables are associated with the HSPS as well.

Technical Reference(s): TQ-TM-104-644-C001 (p13; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the physical connections and/or cause-effect relationships between the EFW and the MFW System. This is accomplished by requiring that the operator identify the number of bistables used in the MFW Pump monitoring circuit, and the parameter that is sensed to determine that the MFW Pumps

are tripped to automatically start the EFW Pumps.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. How many bistables does the MFW Pump monitoring circuit use to input into the HSPS? 2. What parameter do the bistables sense?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	006	K2.02	
	Importance Rating	2.5		

Knowledge of bus power supplies to the following: Valve operators for accumulators

Proposed Question: RO Question # 3

The plant is operating at 100% power.

Which ONE (1) of the following correctly completes the statement below?

1B Core Flood Tank Discharge Isolation Valve CF-V-1B is powered from ____(1)____, and under the current plant conditions the electrical breaker is ____(2)____.

A. (1) 1B ESV MCC
(2) OPEN

B. (1) 1C ESV MCC
(2) OPEN

C. (1) 1B ESV MCC
(2) CLOSED

D. (1) 1C ESV MCC
(2) CLOSED

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-740 (p7; Rev 5), there are three ES related Valve MCCs, 1A ESV MCC, 1B ESV MCC, and 1C ESV MCC; and the operator may incorrectly believe that the 1A CFT Discharge Isolation Valve is powered from 1A ESV MCC, and the 1B CFT Discharge Isolation Valve is powered from 1B ESV MCC.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-213-C001 (p18; Rev 4), CF-V-1A and CF-V-1B are powered from 1C ES Valves MCC. The 1C ES Valves MCC breaker is open and de-energized when the reactor is critical.

C. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the valves automatically open on 500# ESAS.

Technical Reference(s): TQ-TM-104-213-C001 (p18; Rev 4)
TQ-TM-104-740 (p7; Rev 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 213-GLO-4 (As available)

Question Source: Bank # WTSI 60239
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to the valve operators for accumulators. This is done by identifying two bits of information, the Bus supply itself, and how the breaker is administratively controlled.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What is the Bus power supply for CF-V-1B? 2. What is the Breaker's position at 100% power.

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

Knowledge of bus power supplies to the following: Reactor building closed cooling water

Proposed Question: RO Question # 4

Plant conditions:

- 100% power.
- MU-P-1A is operating.
- MU-P-1B is Out of Service.

Which ONE (1) of the following identifies the Intermediate Closed Cooling Pump, and its power supply, that should be operating, AND the reason why?

A. IC-P-1A; (1A ES MCC)
The IC Pump must be powered from the same train as the MU Pump for loss of DC power concerns.

B. IC-P-1A; (1C ES MCC selected to 1S 480V supply)
The IC Pump must be powered from the opposite train as the MU Pump to protect against loss of Reactor Coolant Pump Seal cooling.

C. IC-P-1B; (1C ES MCC selected to 1P 480V supply)
The IC Pump must be powered from the same train as the MU Pump for loss of DC power concerns.

D. IC-P-1B; (1B ES MCC)
The IC Pump must be powered from the opposite train as the MU Pump to protect against loss of Reactor Coolant Pump Seal cooling.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may not know the power supply for IC-P-1A or the basis for alignment, confusion with step 3.3.5 which ensures that DC will be aligned to the same side as seal injection to protect HPI on a loss of one side DC (failure mode MU-V-18).

B. **Incorrect.** This is plausible because the 1C ES MCC powers other components to allow cross over power. The operator may not know the power supply for IC-P-1A but does understand the basis for why an alignment is specified.

C. **Incorrect.** This is plausible because the 1C ES MCC powers other components to allow cross over power. The operator may know the power supply for IC-P-1A and confuse the basis, if they believed that the same train step 3.3.5 applied then this would be logical for the incorrect power supply chosen.

D. **Correct.** According to 1107-5 (p68; Rev 140) the power supply to IC-P-1B is 1B ES MCC. According to OP-TM-211-437, Supplying Seal Injection from MU-P-1A rev 0 step 4.4 initiate OP-TM-541-438 to remove IC-P-1A from service (to start IC-P-1B and place IC-P-1A in standby, the basis of this is found in OP-TM-211-000 rev 24 L&P 2.2.10.

Technical Reference(s): OP-TM-211-437 Rev 0
OP-TM-211-000 Rev 24
1107-5 (p68; Rev 140) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 541-GLO-4 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to the Intermediate closed cooling water (IC)pumps

The question is at the Memory cognitive level because the operator must demonstrate the knowledge of the power supplies for different pumps and the interrelated effect of a loss of power supply.

What MUST be known:
1. The power supply to IC-P-1A and B. 2. The power supply to MU-P-1A. 3. The basis for a separation of trains power supply in this case.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 K3.01	
	Importance Rating	3.8	

Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: Systems controlled by automatic loader

Proposed Question: RO Question # 5

Plant conditions:

- Reactor was tripped from 100% power.
- Small Break LOCA in the RB.
- 1600 psig RCS Pressure ESAS has actuated 2 minutes ago.

Subsequently:

- An undervoltage condition occurs on 4160V Bus 1E.

Assuming that the delay timer in the Emergency Diesel Generator EG-Y-1B Generator Output Breaker is incorrectly set to 5 seconds, which ONE of the following describes the impact on the Block 1 and Block 2 loading with respect to the time of bus under voltage?

A. Block 1 loads will start later than expected;
Block 2 loads will not be impacted.

B. Block 1 loads will start later than expected.
Block 2 loads will start later than expected.

C. Block 1 loads will start sooner than expected;
Block 2 loads will not be impacted.

D. Block 1 loads will start sooner than expected.
Block 2 loads will start sooner than expected.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the Block

2 loads will load on based on the time from the UV condition and not the EDG Output Breaker Closing.

B. **Correct.** According to TQ-TM-104-642-C001 (p40-41; Rev 5), during an event in which the ESAS occurs, and then is subsequently followed by an LOOP, the UV condition starts a 2.5 second timer in the closing circuit of the EDG. The purpose of the 2.5 second timer is to allow the electro-magnetic field of Block 1 motors to collapse prior to the Diesel Generator breaker closing. After 2.5 seconds, the timer times out, Diesel Generator Output Breaker closing re-initiates Block Loading. According to TQ-TM-104-740-C001 (p23; Rev 5), normal Block Loading is initiated on the closure of the EDG Output Breaker. At that time, Block 1 load breakers will close and re-energize their motors, and Block 2 loads will sequence five seconds later. In this event the UV will trip all equipment except for Block 1 load loads off of 4160V Bus 1E. Because the timer is set incorrectly, the EDG Output Breaker will wait five seconds rather than 2.5 seconds before it closes and re-energizes the Block 1 loads. Consequently, the Block 1 loads will start later than they would have if the timer was set correctly. Since Block 2 loads will close in 5 seconds after the EDG Output Breaker closes, these loads will also start later than they would if the timer had been set correctly.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the breaker timer is normally set for 10 seconds, and if so, the breaker would close sooner than expected and re-energize Block 1 loads sooner than expected; and that the Block 2 loads will load on based on the time from the UV condition and not the EDG Output Breaker Closing.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the breaker timer is normally set for 10 seconds, and if so, the breaker would close sooner than expected, and both Block 1 and Block 2 loads would start sooner.

Technical Reference(s):	TQ-TM-104-642-C001 (p40-41; Rev 5) TQ-TM-104-740-C001 (p23; Rev 5)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-5, 740-GLO-5 (As available)

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

Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	7	
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect that a loss or malfunction of the ED/G system (i.e. an improperly set timer on the EDG Output Breaker) will have on the systems controlled by automatic loader.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and apply that information to a failure condition to predict an outcome, in order to correctly answer the question.

What MUST be known:
1. What is the normal time delay in the closing circuit of the EDG Output Breaker? 2. What is the purpose of the time delay of 2.5 seconds in the EDG Output Breaker? 3. How does block loading work? 4. Given the plant conditions, how will Block 1 loads be affected? 5. Given the plant conditions, how will Block 2 loads be affected?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	013	K3.03	
	Importance Rating	4.3		

Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Containment

Proposed Question: RO Question # 6

The following plant conditions exist:

- The plant is at 100% power.
- An inadvertent "B" train 30# ESAS actuation has occurred.
- The "B" train 30# ESAS actuation cannot be reset.

Which ONE (1) of the following are the direct consequences of this ESAS actuation?

A. High Reactor Coolant Pump Seal temperatures;
High Control Rod Drive Stator temperatures.

B. High Reactor Coolant Pump Seal temperatures;
Loss of Reactor Coolant Pump Seal Injection.

C. High Reactor Coolant Pump Motor temperatures;
High Control Rod Drive Stator temperatures.

D. High Reactor Coolant Pump Motor temperatures;
Loss of Reactor Coolant Pump Seal Injection.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because CRDs and RCP thermal barrier heat exchangers lose cooling water due to containment isolation of ICCW, however RCP seal injection will prevent high RCP seal temperatures.

B. **Incorrect.** This is plausible because RCP thermal barrier heat exchangers lose cooling water due to containment isolation of ICCW, but RCP seal injection remains in service.

C. **Correct.** According to 1105-3 (p16; Rev 51), an actuation of the 30# Containment ESAS will result in ICCW and NSCCW being isolated to the Containment. Since Train

B cannot be reset, IC-V-2 (ICCW Inside RB Outlet Valve) and NS-V-35 (Reactor Building Inside component cooling return isolation valve) cannot be re-opened, and therefore neither system can be restored.

D. **Incorrect.** This is plausible because RCP motor coolers and RCP thermal barrier heat exchangers lose cooling water due to containment isolation of NSCCW and ICCW, however RCP seal injection prevents high seal temperatures.

Technical Reference(s): 1105-3 (p16; Rev 51) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-5 (As available)

Question Source:	Bank #	QR-531-GLO-5-Q02	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2000

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate Knowledge of the effect that a malfunction of the ESFAS (i.e. Train B will not reset) will have on the Containment. This is done by requiring the operator to identify the effect on the plant of a failure of Train B ESAS to reset after actuation.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information and then apply what is known to plant conditions to arrive at the correct answer.

What MUST be known:
1. What automatic actions occur on the 30# ESAS actuation? 2. What is the effect of the Train B of 30# ESAS failure to reset?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	063	K4.02	
	Importance Rating	2.9		

Knowledge of dc electrical system design feature(s) and/or interlock(s) which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties.

Proposed Question: RO Question # 7

The plant is operating at 100% power when a Train A 1600 psig ESAS actuation occurs.

Which ONE (1) of the following identifies an action that is prevented from occurring by interlock?

A. Transfer of 1M DC panel to 'B' DC on loss of the 'A' DC Distribution.

B. Transfer of the 1G 480V Bus from the 1B 4160V Bus to the 1N 480V Bus.

C. Transfer of the 6900 Volt Bus 1A to the B Auxiliary Transformer on loss of the A Auxiliary Transformer.

D. Transfer of the 1B Emergency Diesel Generator Breaker to the EMERGENCY position for cooldown outside the Control Room.

Proposed Answer: A

Explanation (Optional):

A. **Correct**. According to OPM A-03 (p4; Rev 12) the 1M DC Bus has an auto transfer switch to auto swap power supplies on a loss of power to the selected bus; DC Panel 1A or 1B. According to 1107-2C (p3; Rev 10) PP&L D, when the discharge cross-connect valves between Makeup Pump MU-P-1B and MU-P-1C are closed, as is the normal 100% lineup, the 1M DC Distribution Panel should be powered from DC Distribution Panel 1A. According to OP-TM-104-642-C001 (p41 and 57; Rev 5), an ES actuation blocks the automatic transfer switch for DC bus 1M.

B. **Incorrect.** This is plausible because according to TQ-TM-104-740-C001 (p21; Rev 5), a feeder is provided between 1D 4160V Switchgear and non-safety related bus 1N BOP 480V bus. An undervoltage on 1D 4160V trips 1N Bus feeder and there is no auto-closure. If the A diesel generator breaker is closed (repowering 1D 4160V bus) and an ES signal exists, then the breaker for 1N bus cannot be re-closed by interlock. Four

other 480V BOP buses can be fed from the 1N 480V bus. The operator may incorrectly believe that the feeder to 1N (N1-02) is opened on the 1600# ESAS, and that a cross-tie between Bus 1G and 1N cannot be made. According to 1107-1 (p61; Rev 81) this would be true if the EDG was supplying the 1D 4160V Bus.

C. **Incorrect.** This is plausible because according to TQ-TM-104-731-C001 (p33-34; Rev 3), interlocks exist to allow an auto transfer of the 6900V Bus power supply if a transformer fault or a substation problem occurs. The scheme is NOT affected by ESAS. The operator may incorrectly believe that the transfer scheme becomes inoperable during an ESAS.

D. **Incorrect.** This is plausible because according to OP-TM-EOP-020, Attachment 9 (p77; Rev 12), Step 1.2 EG-Y-1B is running when the switch positioning is made. The operator may incorrectly believe that the transfer is prohibited when the Diesel is running.

Technical Reference(s):	OPM A-03 (p4; Rev 12) 1107-2C (p3; Rev 10) TQ-TM-104-740-C001 (p21; Rev 5) 1107-1 (p61; Rev 81) TQ-TM-104-731-C001 (p33-34; Rev 3) OP-TM-EOP-020, Attachment 9 (p77; Rev 12)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	734-GLO-5 and 8	(As available)

Question Source:	Bank #	WTSI 65146/IR-734-GLO-8-Q04	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	None

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

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of dc electrical system design feature(s) which provide for breaker interlocks, specifically that an ES actuation will prevent the auto transfer of the 1M Bus power supply.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What is the normal power supply to 1M DC Panel at 100% power (Normally aligned to A). 2. An ES actuation blocks the automatic transfer switch for DC bus 1M.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	008	K4.01	
	Importance Rating	3.1		

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump

Proposed Question: RO Question # 8

Plant Conditions:

- The plant is operating at 100% power.
- Nuclear Services Closed Cooling Water Pumps NS-P-1A and NS-P-1C are ES selected.
- Nuclear Services Closed Cooling Water Pumps NS-P-1A and NS-P-1B are running.

Event Occurrence:

- SBLOCA occurs.
- RCS pressure is 1500 psig and lowering slowly.
- RB pressure is 5 psig and rising slowly.

Which ONE (1) of the following identifies the status of the Nuclear Service Component Cooling Water System?

A. NS-P-1A continues to run;
NS-P-1B continues to run; AND
NS-P-1C starts after a 10-second time delay.

B. NS-P-1A continues to run;
NS-P-1B trips; AND
NS-P-1C starts immediately.

C. NS-P-1A trips and restarts after a 10 second time delay;
NS-P-1B trips; AND
NS-P-1C starts after a 10 second time delay.

D. NS-P-1A trips and restarts after a 10 second time delay;
NS-P-1B continues to run; AND
NS-P-1C starts immediately.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to TQ-TM-104-642-C001 (p28-29; Rev 5), the High Pressure Injection Engineered Safeguards actuation signal (ESAS) occurs when Reactor Coolant System pressure is at 1600 psig on 2 of 3 bistables, and the 4 psig Engineered Safeguards actuation signal (ESAS) occurs when Reactor Building pressure is at 4 psig on 2 out of 3 bistables. Consequently an ESAS signal has been generated. According to TQ-TM-104-642-C001 (p40; Rev 5), Block 1 loads actuate upon Engineered Safeguard signal initiation, and Block 3 loads actuate ten seconds after Block 1. According to TQ-TM-104-531-C001 (p38; Rev 6), the ES selected NS pumps start on Block 3, and unlike the NR pumps there are no ESAS trip signals to any of these pumps unless there is an undervoltage condition on the ES Bus. According to 1105-3 Attachment 1 (p8; Rev 51), the NS Pumps are tripped and locked out when an ESAS occurs with an undervoltage condition on the ERS Bus. Therefore, since NS-P-1A and NS-P-1C are ES selected, NS-P-1A, which does not receive a trip signal, continues to run (it is advantageous to continue running and the ESAS will not penalize plant operation). Additionally, NS-P-1C will start 10 seconds after the ESAS (Block 3). Also, since NS-P-1B was running at the time of the ESAS, and there is no undervoltage condition on the ES Bus, it will continue to run.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the non-selected NS Pump would receive a trip signal; AND incorrectly believe that the non-running but ES-selected NS Pump will start immediately. According to TQ-TM-104-531-C001 (p37; Rev 6) the Non-ES Selected NR pump will trip on an ES actuation, and the operator may confuse the operation of the NS and NR Pumps.

C. **Incorrect.** This is plausible because the operator may not understand the operation of a running and ES-Selected pump; incorrectly believing that the running and ES-Selected pump will respond to Block 3 and load shed and restart. Additionally, the operator may incorrectly believe that the non-selected NS Pump would receive a trip signal.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the non-running but ES-selected NS Pump will start immediately; AND incorrectly believe that the running and ES-Selected pump will respond to Block 3 and load shed and restart.

Technical Reference(s):	TQ-TM-104-642-C001 (p28-29, 40; Rev 5) TQ-TM-104-531-C001 (p38; Rev 6) 1105-3 Attachment 1 (p8; Rev 51)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	541-GLO-10		(As available)
Question Source:	Bank #	QR-541-GLO-10-Q02	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	None
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	7	
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of CCWS design feature(s) and/or interlock(s) which provide for the automatic start of standby pump

The question is at the Comprehension/Analysis cognitive level because the operator must use knowledge of NS Pump starts on ESAS means and manner, and evaluate plant conditions, to determine the correct answer.

What MUST be known:
1. How are the NS Pumps started on an ESAS (i.e. Block 1 v. Block 3)? 2. How does an operating NS Pump respond to an ESAS signal (i.e. load shed and restart v. continue running)? 3. How does a non-operating NS Pump respond to an ESAS signal (i.e. start immediately v. Block loading)? 4. Does the non-ES selected NS Pump receive a trip signal on an ESAS?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	007	K5.02	
	Importance Rating	3.1		

Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR

Proposed Question: RO Question # 9

Plant conditions:

- 1103-11, RCS Water Level Control, is controlling procedure preparing for RCS heatup.
- RCS final fill operation has been completed, with flow into the RCS terminated.
- Pressurizer level is 390 inches.
- Pressurizer temperature is 190 degrees F.
- Reactor Coolant Drain Tank (RCDT) pressure is 2.0 psig.
- Operator energizes Pressurizer heaters to form a steam bubble in the Pressurizer.

Event, 60 minutes later:

- RCS Pressure indication reaches 22 psig.
- RCDT level begins to rise.

Based on these conditions, which ONE (1) of the following describes the source of the water flowing into the RCDT?

A. Hot leg vent(s)

B. Pressurizer vent

C. RCP Seal Standpipe(s)

D. Center Control Rod Drive mechanism vent

Proposed Answer: A

Explanation (Optional):

A. **Correct.** RCDT will begin to rise when pressure is high enough to overfill one or both hot legs (flow will initiate out the hot leg vent(s). According to Steps 3.3.2.11-15 of

1103-11 (p17-21, 49 Rev 68), when Pzr level is 390 ±2 inches RCS fill is terminated, and the manual Pzr vent valves are closed. When Pzr temperature has been > 230F for > 30 minutes, the Pzr Vent valves to the RCDT Sparger (RC-V-28 and 44) are closed. With a steam bubble in the Pzr, maintaining RCS pressure at 22 psig water will be forced out of the Pzr and into the RCS causing water to issue from the Hot Leg vents (RC-V15A/B, RC-V46A/B and RC-V-14A/B) and into the RDCT. A note is provided indicating that RCDT level should rise when RCS Pressure (RC3A-PT5) is between 20 and 24 psig. At this point a Steam Bubble exists in the Pzr.

B. **Incorrect.** This is plausible because the Pzr vents to the RCDT, and the Pzr manual vent valves are open during the RCS filling process, even at the point that the Pzr heaters are energized (Steps 3.3.2.10-11 of 1103-11).

C. **Incorrect.** This is plausible because the RCP Seal standpipes overflow and drain to the RCDT. However, flow from the RCP standpipes is not possible at these conditions, and the standpipe bypass valves, RC-V-33A-D, are closed in accordance with OP-TM-220-000 (p33; Rev 13).

D. **Incorrect.** This is plausible because the CRD venting system vents to the RCDT. However, In accordance with 1103-11, Enclosure 3A, (p5of 6; Rev 68), the CRDM are vented and closed when the RCS is being filled, and Pzr level is between 280-360 inches. This would have already occurred at this point in the procedure.

Technical Reference(s): 1103-11 (p17-21, 49 Rev 68)
1103-11, Enclosure 3A, (p5of 6; Rev 68)
OP-TM-220-000 (p33; Rev 13)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-12-PCO4 (As available)

Question Source: Bank # IR-GOP-012-PCO-4-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge (i.e. indications of a steam bubble in the Pzr such as vent valves closed, 20-25 psig in the RCS and water issuing from the high point vents) of the operational implications of the method of forming a steam bubble in the PZR as it applies to PRTS.

The question is at the Comprehension/Analysis cognitive level because the operator must understand the process of drawing a bubble in the Pzr (i.e. with water level above the heaters but below the top of the Pzr, the heaters are energized, the water is brought to saturation and boiling occurs, and then the Pzr Vents are closed), and then relate it to its consequence (i.e. once the vents are close, water will back up into the system and ultimately issue from the high point, which must be known as well).

What MUST be known:
1. When are the CRDM vent valves closed during RCS fill? 2. Under the stated plant conditions what is the position of the RCP standpipe bypass valves, RC-V-33A-D. 3. When are the Pzr Vent valves closed during the process of drawing a steam bubble in the Pzr? 4. What is the cause of inflow into the RCDT during the process of drawing a steam bubble in the Pzr? 5. What action is taken when RCS pressure is between 20-24 psig and RDCT Level starts to rise during the process of drawing a steam bubble in the Pzr?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K5.01	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables

Proposed Question: RO Question # 10

The plant has been stabilized following an overcooling transient and the following conditions exist:

- Tavg is 550°F.
- RCS Pressure is 2010 psig.
- Pressurizer Level is 250 inches and slowly increasing.
- Pressurizer Temperature is 610°F.
- All Pressurizer Heaters are energized.

Which ONE (1) of the following correctly completes the statement below?

The pressurizer is ____(1)____ for the current RCS pressure and the ____(2)____ maintaining RCS pressure.

A. (1) saturated
(2) Pressurizer Heaters are

B. (1) subcooled
(2) Pressurizer Heaters are

C. (1) saturated
(2) compressed steam bubble in the pressurizer is

D. (1) subcooled
(2) compressed steam bubble in the pressurizer is

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the Pressurizer Heaters are controlling pressure, which would be the case if the plant was solid (along with makeup/letdown flow, or mass input/output).

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly use the Steam Tables and determine that the Pressurizer fluid is saturated.

D. **Correct.** 1st part correct, 2nd part correct. Using Steam Tables, the operator will compare the RCS Pressure of 2010 psig (or 2025 psia) against Table 2, Saturated Steam Pressure, and determine that the associated saturated temperature is 637°F. Since the Pressurizer fluid temperature is 610°F, the Pressurizer fluid is 27°F subcooled. According to TQ-TM-104-220-C001 (p21-22; Rev 5), the volume of the steam space is controlling Pressurizer level under normal and transient conditions.

Technical Reference(s): Steam Tables, Table 2 TQ-TM-104-220-C001 (p21-22; Rev 5) (Attach if not previously provided)

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

Learning Objective: 220-GLO-2 (As available)

Question Source: Bank # WTSI 60001

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2004

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications and determination of the condition of fluid in PZR, using steam tables. This is accomplished by not only requiring that the operator determine the fluid condition for a given set of conditions, but by also identifying how the RCS pressure is being controlled in those conditions.

The question is at the Comprehension/Analysis cognitive level because the operator must apply the use of Steam Tables to correctly answer the first part of the question, and then demonstrate an understanding how the Pressurizer operates while at power to answer the second part of the question correctly.

What MUST be known:
1. How to use Steam Tables to determine the condition of the water in the Pressurizer at any given time. 2. How the pressurizer operates to maintain RCS pressure while the plant is at power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	K6.17	
	Importance Rating	4.4		

Knowledge of the operational implications of the following concepts as they apply to the CVCS: Flow paths for emergency boration

Proposed Question: RO Question # 11

Plant conditions:

- Emergency Boration is required.
- MU-V-14A and MU-V-14B, Makeup pump suction from BWST, are stuck closed.
- Guide 1 method "B" is being used for backup emergency boration.
- Boric Acid Mix Tank is out of service.
- WDL-T-7A Reclaim Boric Acid Tank (RBAT) is on recirculation.

Which ONE (1) of the following correctly completes the statement below?

The flow path from WDL-T-7A RBAT to the Makeup tank is through...

A. WDL-V-61 Boric Acid Injection Valve, and the batch controller.

B. WDL-V-61 Boric Acid Injection Valve, and bypassing the batch controller.

C. MU-V-51 Emergency Boric Acid Addition Valve, and the batch controller.

D. MU-V-51 Emergency Boric Acid Addition Valve, and bypassing the batch controller.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-EOP-010 Guide 1 method "B" Position WDL-V-61 to inject, Maximize the batch size, Open MU-V-10, (next steps ensure 1 RBAT is on Recirc.

B. **Incorrect.** This is plausible because method "A" using MU-V-51 bypasses the batch controller, incorrect Guide 1 directs using WDL-V-61 and the use of the batch controller.

C. **Incorrect.** This is plausible because method "A" (BAMT) using MU-V-51 bypasses the

batch controller, incorrect Guide 1 directs the use of the batch controller.

D. **Incorrect.** This is plausible for confusing Method "A" (BAMT) with flow path for method "B" (RBAT).

Technical Reference(s): OP-TM-EOP-010 (p13; Rev 12) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-12 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of the flow paths for emergency boration, flow path and whether or not the batch controller is required to be used.

The question is at the Memory cognitive level because the operator must recall two pieces of information which valve and the subsequent flow path.

What MUST be known:
1. Guide 1 method "B" path MU-V-51 or WDL-V-61 which path is associated with RBATS. 2. WDL-V-61 flow path whether it goes directly to the inlet of the MU tank or through the batch controller.

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

Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger

Proposed Question: RO Question # 12

Which ONE (1) of the following events would prevent Simultaneous DHR and LPI/HPI Operation initiation in accordance with OP-TM-212-921, Simultaneous DHR and LPI/HPI Operation, following an RCS LOCA?

A. Loss of Instrument Air.

B. Loss of Vital bus "B" (VBB).

C. All Reactor Coolant Pumps tripped.

D. DR Pump Discharge Valve DR-V-1B is stuck closed.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible since DC-V-2B and DC-V-65B valves will fail on loss of instrument air; however they fail to their ESAS positions.

B. **Incorrect.** This is plausible since DC-V-2B and DC-V-65B valves will fail on loss of Vital bus "B" however they fail in their ESAS positions.

C. **Incorrect.** This is plausible since different RCS pressure values will be in effect if an RCP is operating; however the procedure can be initiated in either case.

D. **Correct.** According to OP-TM-212-921 (p2; Rev 3), the operator must verify that both Decay Heat river water trains are OPERABLE prior to implementing this procedure. Since Decay River Water is not available to the B DHR heat exchanger the prerequisite is not met.

Technical Reference(s): OP-TM-212-921 (p2; Rev 3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-12 (As available)

Question Source: Bank # IR-212-GLO-12-Q05

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect of a loss or malfunction on the RHR heat exchanger will have on the RHRS. Specifically, the operator must know that post-LOCA in which simultaneous DHR and LPI/HPI operation is required, if the B DR Train is not available for operation, simultaneous DHR and LPI/HPI operation is not permitted.

The question is at the Comprehension/Analysis cognitive level because the operator must utilize knowledge of the normal ES Standby alignment, and then evaluate the consequences of four different sets of conditions, and identify the one that will prohibit simultaneous DHR and LPI/HPI operations, to correctly answer the question.

What MUST be known:
1. Both DR trains must be OPERABLE in order to perform OP-TM-212-921, Simultaneous DHR and LPI/HPI Operation. 2. A Loss of Instrument Air will not render the DHR HX inoperable on ESAS. 3. A Loss of VBB will not render the DHR HX inoperable on ESAS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 A1.02	
	Importance Rating	2.9	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP pump and motor bearing temperatures

Proposed Question: RO Question # 13

Plant conditions:

- 100% power.
- Total loss of Nuclear Services Closed Cooling Water occurred 5 minutes ago.
- Attempts to start NS-P-1A/B/C were unsuccessful.

Assuming that all temperatures are rising at one degree per minute (°C or °F as applicable), which ONE (1) of the following conditions will require the operator to shut down the Reactor Coolant Pumps within the next 60 seconds?

A. Motor stator temperature indication is 135°C.

B. Motor guide bearing temperature indication is 165°F.

C. Motor thrust bearing temperature indication is 200°F.

D. #1 Seal water radial bearing temperature indication is 220°F.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-031 (p1 and 3; Rev 4) the RCPs must be tripped if the RCP motor or bearing temperature is approaching HI-2 alarm set point. However, according to OP-TM-PPC-A0704 (p1; Rev 1), the RCP Stator Temperature Hi-1 alarm occurs at 125°F, and the Hi-2 alarm occurs at 150°F. The Hi-2 alarm setpoint has not been reached.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-031 (p1 and 3; Rev 4) the RCPs must be tripped if the RCP motor or bearing temperature is approaching HI-2 alarm set point. However, According to OP-TM-PPC-A0702 (p1; Rev 2), the RCP Motor

Upper Guide Bearing Temperature Hi-1 alarm occurs at 165°F, and the Hi-2 alarm occurs at 185°F, and according to OP-TM-PPC-A0703 (p1; Rev 1), the RCP Motor Lower Guide Bearing Temperature Hi-1 alarm occurs at 165°F, and the Hi-2 alarm occurs at 185°F. Although the Hi-1 alarm setpoint has been exceeded for both upper and lower bearings, at the present rate of rise, the Hi-2 setpoint is still five minutes away.

C. **Correct.** According to OP-TM-AOP-031 (p1 and 3; Rev 4) identifies an entry condition as Less than two NS pumps operating and RCP motor or bearing temperature approaching HI-2 alarm set point. Upon entry into the procedure, the operator will be directed to trip the reactor and stop all RCPs. According to OP-TM-PPC-A0700 (p1; Rev 2), the RCP Motor Down Thrust Bearing Temperature Hi-1 alarm occurs at 175°F, and the Hi-2 alarm occurs at 200°F. According to OP-TM-PPC-A0701 (p1; Rev 2), the RCP Motor Up Thrust Bearing Temperature Hi-1 alarm occurs at 190°F, and the Hi-2 alarm occurs at 200°F. Since the Hi-2 alarm has been exceeded in both instances, a trip of all RCPs is required.

D. **Incorrect.** This is plausible because according to OP-TM-AOP-031 (p1 and 3; Rev 4) the RCPs must be tripped if the RCP motor or bearing temperature is approaching HI-2 alarm set point. However, according to OP-TM-PPC-A0521 (p1; Rev 1), the RCP Seal Water Temperature Hi-1 alarm occurs at 190°F, and the Hi-2 alarm occurs at 225°F. Although the Hi-1 alarm setpoint has been exceeded, at the present rate of rise, the Hi-2 setpoint is still five minutes away.

Technical Reference(s): OP-TM-AOP-031 (p1 and 3; Rev 4)
OP-TM-PPC-A0700 (p1; Rev 2)
OP-TM-PPC-A0701 (p1; Rev 2)
OP-TM-PPC-A0699 (p1; Rev 1)
OP-TM-PPC-A0702 (p1; Rev 1)
OP-TM-PPC-A0703 (p1; Rev 1)
OP-TM-PPC-A0521 (p1; Rev 1)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-9 (As available)

Question Source: Bank # IR-226-GLO-9-Q02

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	1998

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate the ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCP pump and motor bearing temperatures.

The question is at the Memory cognitive level because the operator must recall several bits of information to correctly answer the question.

What MUST be known:
1. The criteria for RCP shutdown in AOP-031 is RCP motor or bearing temperature approaching HI-2 alarm set point. 2. The Hi-2 Alarm setpoint for the RCP Motor temperature. 3. The Hi-2 Alarm setpoint for the RCP Upper Motor Bearing temperature. 4. The Hi-2 Alarm setpoint for the RCP Lower Motor Bearing temperature. 5. The Hi-2 Alarm setpoint for the RCP Motor Down Thrust Bearing temperature. 6. The Hi-2 Alarm setpoint for the RCP Motor Up Thrust Bearing temperature. 7. The Hi-2 Alarm setpoint for the RCP Seal Water at Radial Bearing temperature.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A1.01	
	Importance Rating	3.6	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment temperature

Proposed Question: RO Question # 14

Which ONE (1) of the following conditions, if it existed as stated, would violate a Containment temperature limitation as identified in OP-TM-823-000, Reactor Building Heating and Ventilation System?

A. With Containment Integrity required, and RB Purge in progress, the RB temperature below the 320' elevation level is 98°F.

B. With Containment Integrity required, and a plant heatup in progress, the RB temperature below the 320' elevation level is 68°F.

C. With Containment Integrity NOT required, and RB Purge in progress, the RB temperature below the 320' elevation level is 58°F.

D. With Containment Integrity NOT required, and RB Purge in progress, the RB temperature above the 320' elevation level is 128°F.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-823-000 (4; Rev 6) Step 2.2.2, the operator is directed to maintain purge supply (AH-TI-6A & B) and purge exhaust (average RB temperature below 320' elev. or locally measured ambient temperature near purge valves) ≥90°F when purging while containment integrity is required. The operator may not know the limit.

B. **Correct.** According to OP-TM-823-000 (4; Rev 6) Step 2.2.3, the operator is directed to maintain RB average temperature below 320' elevation above 70°F in order to minimize the probability of OTSG TSDT problems during plant heatup. Since a plant heatup is in progress RB temperature must be at least 70°F.

C. **Incorrect.** This is plausible because according to OP-TM-823-000 (4; Rev 6) Step 2.2.2, the operator is directed to maintain purge supply (AH-TI-6A & B) and purge

exhaust (average RB temperature <u>below</u> 320' elev. or locally measured ambient temperature near purge valves) ≥55°F when purging while containment integrity is not required. The operator may not know the limit.

D. **Incorrect.** This is plausible because according to OP-TM-823-000 (4; Rev 6) Step 2.2.5, the Maximum allowable RB average temperature above 320' elevation is 130°F. The operator may not know the limit.

Technical Reference(s): OP-TM-823-000 (p4; Rev 6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 823-GLO-9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor changes in parameters (to prevent exceeding design limits) associated with containment temperature. The ability to monitor changes is demonstrated by knowing where the limit lies (i.e. above or below the 320' elevation), and the ability to prevent exceeding design limits is demonstrated in knowing the limits.

The question is at the Memory cognitive level because the operator must simply recall the established limit on containment temperature, given a plant condition.

What MUST be known:
1. What is the RB Temperature limit below elevation 320' when a heatup is in progress.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103	A2.05	
	Importance Rating	2.9		

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 or operations Emergency containment entry

Proposed Question: RO Question # 15

Plant conditions:

- Plant shutdown from power operation is in progress.
- A Train of DHR is in service.
- RCS temperature is 194°F.
- RCS pressure is 310 psig.

Event:

- Fire alarms occur inside the Unit 1 Reactor Building, and the crew enters OP-TM-AOP-001, Fire.
- The Shift Manager suspects an active fire and decides to send the Fire Brigade into the Unit 1 Reactor Building.
- The Fire Brigade determines that a water hose must be run through BOTH doors of the Personnel Air Lock.

Which ONE (1) of the following correctly completes the statement below?

When both Airlock doors are open, entry into Technical Specification 3.6.1, Reactor Building – Containment Integrity ____(1)____ required, and this entry will be considered ____(2)____ in accordance with RP-TM-460-1007, Access to TMI-1 Reactor Building.

A. (1) is
(2) a Planned Reactor Building Entry

B. (1) is
(2) an Urgent Unplanned Reactor Building Entry

C. (1) is NOT
(2) a Planned Reactor Building Entry

D. (1) is NOT
(2) an Urgent Unplanned Reactor Building Entry

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and D.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that under the present conditions CONTAINMENT INTEGRITY must be maintained.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may be unfamiliar with the distinguishing characteristics between the Urgent Unplanned and Planned Entry, and incorrectly conclude that this entry is NOT urgent. Additionally, the operator may incorrectly believe that since the plant is no longer at power, an Urgent Entry is not needed.

D. **Correct.** 1st part correct, 2nd part correct. According to Technical Specification 3.6.1 (p3-41; Amendment 246), except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (Section 1.7) shall be maintained whenever *all* three of the following conditions exist: (1) Reactor coolant pressure is 300 psig or greater, (2) Reactor coolant temperature is 200 degrees F or greater, and (3) Nuclear fuel is in the core. Since the RCS temperature is less than 200°F, the maintenance of Containment Integrity is NOT required, and both Personnel Airlock Doors can be opened without entering into TS 3.6.1. According to RP-TM-460-1007 (p6-7; Rev 6) there are two types of Reactor Building Entries made with this procedure; (1) and Urgent Unplanned Entry, and a (2) Planned Entry. The Planned Entry is characterized by RadPro Briefings which will require signoffs. The Urgent Unplanned Entry is characterized by limited restrictions such as no RWP, a minimum number of persons making the entry, and the bypassing of typical entry requirements. According to OP-TM-AOP-001 (p1-7; Rev 8) Steps 3.1 through 3.18, actions are taken to get the Fire Brigade into the Containment such as making an announcement as to the location, ensure that Security responds to access the area, and aligning the Containment Fire Service to the Containment, all in an effort to effectively and efficiently deal with the fire. The Urgent Unplanned Entry is consistent with this response.

Technical Reference(s): Technical Specification 3.6.1 (p3-41; Amendment 246)
RP-TM-460-1007 (p6-7; Rev 6)
OP-TM-AOP-001 (p1-7; Rev 8)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-14 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to (a) predict the impacts of the emergency containment entry on the containment system (Log into TS, or no), and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those operations. This is accomplished by presenting a situation involving an Emergency Containment Entry that will relax Containment Integrity; and then requiring that the operator identify whether or NOT the Containment Integrity Technical Specification will need to be entered; and then require that the operator decide upon the implementation of two proceduralized entry paths.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. When the RCS is not open to the atmosphere, what conditions must exist to require Containment Integrity to be maintained? 2. In the given conditions, is Containment Integrity required to be maintained? 3. When the Fire Brigade enters the Containment during a fire event, how will they enter the Containment?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026	A2.04
	Importance Rating	3.9	

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of spray pump

Proposed Question: RO Question # 16

Plant conditions:

- The plant has tripped from 100% power.
- The RCS 1600 psig and 500 psig ESAS has actuated.
- The RB pressure 4 psig and 30 psig ESAS has actuated.

Subsequently:

- MAP-E-2-3, RB SPRAY FLOW LO, alarms.
- A Train BS flow is 750 gpm.
- B Train BS flow is 1100 gpm.

Which ONE (1) of the following correctly assesses the operational status of the RB Spray Pumps, AND identifies the required action?

A. RB Spray Pump BS-P-1A is operating abnormally, ONLY;
Place RB Spray Pump BS-P-1A in PTL within 60 seconds.

B. RB Spray Pump BS-P-1A is operating abnormally, ONLY;
Initiate OP-TM-214-901, RB Spray Operation.

C. RB Spray Pumps BS-P-1A and 1B are operating abnormally;
Initiate OP-TM-214-901, RB Spray Operation.

D. RB Spray Pumps BS-P-1A and 1B are operating abnormally;
Place RB Spray Pump BS-P-1A and BS-P-1B in PTL within 60 seconds and verify that all RB Cooling Units are operating in SLOW speed.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the pump assessment is correct, and according to OP-TM-214-000 (p1; Rev 8), the operator is directed to not operate a BS pump for more than 60 seconds with flow less than 60 gpm. The operator may incorrectly apply the 60 second rule to the low flow condition.

B. **Correct.** According to TQ-TM-104-214-C001 (p22; Rev 6), the minimum required BS Pump flow is 800 gpm, and the design normal flow is 1100 gpm. Therefore, BS-P-1B is operating as expected. According to OP-TM-MAP-E0203 (1; Rev 3), the RB SPRAY FLOW LO alarm will occur when specific train flow is <900 gpm, and therefore, this indicates that BS-P-1A flow is abnormally low. According to OP-TM-MAP-E0203 (1; Rev 3), when the low flow condition exists, and the 30 psig ESAS has actuated the operated is directed to initiate OP-TM-214-901.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the normal flow of a BS Pump is greater than 1100 gpm.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the normal flow of a BS Pump is greater than 1100 gpm. According to OP-TM-214-901 (p6; Rev 4) the operator is directed to monitor the RB Spray system after auto or manual actuation and check for flow > 1400 gpm. If so, the operator is directed to place the BS Pump in PTL. The operator may incorrectly believe that both pumps should be operating at 1400 gpm and that this is the cause of the low flow alarms. If both BS Pumps were turned off the logical action would be to verify alternate cooling.

Technical Reference(s): TQ-TM-104-214-C001 (p22; Rev 6)
OP-TM-MAP-E0203 (1; Rev 3)
OP-TM-214-901 (p6; Rev 4)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 214-GLO-10 and 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate the ability to predict the impact of an RB Spray Pump failure; and identify the required action. This is accomplished by presenting the operator with a set of conditions, and the information that at least one RB Spray Pump is operating abnormally, and requiring the operator to assess pump operation, and then based on the evaluation select the actions to take.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What is the setpoint of the low flow alarm, and the cause of it being in under the present conditions? 2. What is the normal flow of an operating BS Pump under the stated conditions? 3. What action must be taken if the flow in one or more RB Spray trains is abnormally low?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 A3.01	
	Importance Rating	3.1	

Ability to monitor automatic operation of the IAS, including: Air pressure

Proposed Question: RO Question # 17

Plant conditions:

- 100% power.
- Instrument Air has lowered and is currently steady at 81 psig throughout the system.

Assuming no operator action, which ONE (1) of the following identifies the current position of the following Instrument Air System valves?

- IA-V-1, Instrument Air Backup from Service Air.
- IA-V-2104A/B, Instrument Air Auto Isolation Valves.
- IA-V-2133, IA-Q-2 Dryer Bypass Valve.

	IA-V-1	IA-V-2104A/B	IA-V-2133
A.	Closed	Open	Open
B.	Closed	Open	Closed
C.	Open	Closed	Open
D.	Open	Closed	Closed

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the IA System is designed such that the Auto Isolation Valves IA-V-2104A/B, and the IA-Q-2 Bypass Valve are the only valves that will be opened at this pressure, and that the other valve will be opened at a lower pressure.

B. **Correct.** According to PLB-1-7 (p1; Rev 6) on lowering IAS pressure, the following automatic actions will occur: (1) at 85 psig, IA-V-2104A/B open and IA-P-1A/B supply IA headers, (2) at 80 psig, (+0 -5) IA-V-1 opens and (3) At 75 psig, (+0 -5) IA-V-2133 bypass on IA-Q-2 opens. According to TQ-TM-104-850-C001 (p22; Rev 2), the Auto Isolation valves, IA-V-2104A/B, will open when the local Instrument Air header pressure (PS-1404) drops to 85 psig. Consequently this valve is open. According to TQ-TM-104-850-C001 (p88; Rev 2), IA-V-1, Instrument Air backup supply valve from Service Air, opens at 80 psig. Consequently, this valve is closed. According to TQ-TM-104-850-C001 (p35; Rev 2), in Auto, IA-Q-2 Bypass Valve, IA-V-2133, will open (bypass IA-Q-2) when Instrument Air pressure at PS-1406 is <75 psig. Consequently, this valve is closed.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the IA System is designed such that the IA Backup from SA IA-V-1, and the IA-Q-2 Bypass Valve are the only valves that will be opened at this pressure, and that the other valves will be opened at a lower pressure.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the IA System is designed such that the IA Backup from SA IA-V-1 is the only valve that will be opened at this pressure, and that the other valves will be opened at a lower pressure. This is enhanced by the fact that two additional backup methods are available when the pressure drops < 80 psig; and the operator may incorrectly believe that SA IA-V-1 is the one that opens at 81 psig.

Technical Reference(s): PLB-1-7 (p1; Rev 6)
TQ-TM-104-850-C001 (p22, 35 and 88; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	4
	55.43	

Secondary coolant and auxiliary systems that affect the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the IAS, including air pressure. This is accomplished by presenting the operator with a set of conditions which includes a lowering IA pressure, and requiring the operator to conclude whether or not an automatic action has occurred.

The question is at the Comprehension/Analysis cognitive level because the operator must recall the automatic function setpoints of three valves, and then given an IA System pressure, conclude whether or not an automatic action has occurred, to correctly answer the question.

What MUST be known:
1. At what pressure will the IA Backup from SA automatically open on lowering system air pressure? 2. At what pressure will the Auto Isolation Valves automatically open on lowering system air pressure? 3. At what pressure will the IA-Q-2 Bypass Valve automatically open on lowering system air pressure? 4. Under the present plant conditions, what position would these valves be in?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	A3.01	
	Importance Rating	3.0		

Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage

Proposed Question: RO Question # 18

Plant conditions:

- Shut down.
- Maintenance requires 1P 480 volt bus to be fed from 1S 480 volt bus.

In accordance with OP-TM-731-550, Cross Tie of ES 480 Volt Bus Feeding 1P From 1S, which ONE of the following:
(1) Identifies where current to the 1P 480 volt bus is monitored, and
(2) The means by which this current is monitored?

A. (1) 4160 V Line to 1P Bus;
(2) By calculating.

B. (1) 480 V Line to 1P Bus;
(2) By direct reading.

C. (1) 4160 V Line to 1S Bus;
(2) By calculating.

D. (1) 480 V Line to 1S Bus;
(2) By direct reading.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may know that no 480 volt amperage indication exists, but incorrectly believe that the supply breaker from its associated 4160 Volt bus remains closed during crosstie operations.

B. **Incorrect.** This is plausible because the operator may not know that 480V amperage indication is NOT available.

C. **Correct.** Current cannot be monitored directly and is determined from the 4160 V feeding bus. According to OP-TM-731-550 (p2; Rev 1), the operator is directed to verify that the load to be picked up is less than 640 amps, or is within the capability of the Emergency Diesel, if it is the sole provider of power to the supplying 4KV bus. A note provided indicates that 640 amps of load at 480 volts corresponds to 74 amps of load as read on the 4KV bus. Since 480 volt bus amps are not metered in the control room, monitor the amps read on the ammeter for the supplying breaker on Console Right. Subsequently, according to Step 4.1.2 (p3; Rev 1), the operator is directed to record the currents of the 2 busses being tied together, and verify that the total is less than 1500 amps (173 amps high side) for 1P tie with 1S. The operator must then use the ammeter for the supplying breaker (4160 V Line to 1S Bus) to determine the amperage on the supplying bus, then use the ammeter for the supplying breaker (4160 V Line to 1P Bus) to determine the amperage on the bus to be supplied. Calculations are needed to arrive at these values. These two amperages are then added together and compared a maximum amperage limit of 173 amps, which will subsequently be used to establish the limit for the ammeter on the 4160 V Line to 1S Bus, while Bus 1S is cross-tied to Bus 1P.

D. **Incorrect.** This is plausible because the operator may not know that 480V amperage indication is NOT available.

Technical Reference(s): OP-TM-731-550 (p2-3; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10 (As available)

Question Source: Bank # WTSI 58752

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	8
	55.43	

Components, capacity, and functions of emergency systems.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the ac distribution system, specifically Vital ac bus amperage when two vital AC busses are cross-tied.

The question is at the Memory cognitive level because the operator must recall specific facts to correctly answer the question.

What MUST be known:
1. The means of monitoring amperage on 1P. 2. The means of monitoring amperage on 1S. 3. The normal electrical breaker alignment during 480V AC Bus cross-tie operations. 4. The means that OP-TM-731-550 establishes for monitoring maximum amperage limits are not exceeded during 480V AC Bus cross-tie operations.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A4.07	
	Importance Rating	2.8		

Ability to manually operate and/or monitor in the control room: Steam dump valves.

Proposed Question: RO Question # 19

Plant conditions:

- Reactor tripped from 100% power.
- ICS Turbine Bypass Valve/Atmospheric Dump valve control stations are in automatic.
- Condenser vacuum is low at 20 inches due to condenser air leak.
- OTSG pressures are stable at 1010 psig.

Assuming NO operator action, which ONE (1) of the following describes the response of OTSG pressure control systems if the Main Condenser vacuum rises to 25 inches?

A. TBVs open to automatically control OTSG pressures at 1010 psig;
ADVs close, but will open if OTSG pressure exceeds 1040 psig.

B. ADVs will automatically control OTSG pressures between 1026-1052 psig;
TBVs remain closed.

C. ADV control is transferred to the (control room) Backup Manual Loaders;
TBVs remain closed.

D. TBV control is automatically transferred to ICS Manual;
ADVs close, but will open if OTSG pressure exceeds 1040 psig.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible if the operator incorrectly believes that the TBVs are not latched closed until reset.

B. **Correct.** According to TQ-TM-104-411 (p29-31; Rev 5), the TBV/ADV pressure control schemes use OTSG pressure signals. During normal operation the setting is 885 psig,

corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10, 75 and 125, resulting in normal automatic adjustable control setpoints of 895, 960 and 1010 psig. The ICS has fixed Automatic Control setpoints as well. Fixed ICS setpoints of 1040 psig and 1026 psig can be applied (depending on plant conditions) in order to limit high OTSG pressures. The 895 psig Automatic Control setpoint provides a 10 psi control band allowance to prevent Turbine Bypass Valves and Turbine Control Valves from fighting each other during Turbine startup and low load operation when both flow paths are in operation at the same time. The 960 psig Automatic Control setpoint is in effect, when ICS ULD > 15%. This setpoint is intended to prevent inadvertent Turbine Bypass Valve operation during normal plant transients. The 1010 psig Automatic Control setpoint is to prevent excessive Pressurizer Level decrease on reactor trip. Raising the OTSG pressure setpoint on a reactor trip limits RCS Cooldown and shrink, since this sets post trip OTSG saturated steam temperature at 555°F. The 1040 psig Automatic (Fixed) Control setpoint provides an independent high-pressure relief that will open proportional to OTSG pressure. The circuit will transfer to MS-V-4A/4B controls on low vacuum or loss of CW Pumps. The 1026-1052 psig Automatic (Fixed) Proportional Control setpoint will cause the ADVs to modulate open proportionally to control steam flow as a function of pressure in the range 1026 to 1052 psig. This circuit is never used by the TBVs, and is blocked (not used) by the ADVs if Main Condenser vacuum is <23" Hg or less than 2 CW Pumps are running. The TBV/ADV automatic transfer from the Turbine Bypass Valves to the Atmospheric Dump Valves on loss of vacuum, (23" Hg) or less than two Circulating Water Pumps operating is sensed by the CVI relay. When this occurs, ICS sends the TBVs a 0% demand signal. Under either of these two conditions, MS-V-4A/4B (if in automatic) will control pressure at the operator controlled ICS setpoint (normally 885 psig) + bias, or at 1040 psig (fixed setpoint), whichever error signal is greater. Additionally, MS-V-4A/4B will open at 1026 psig and be fully open at 1052 psig to protect OTSGs from overpressure. This signal is blocked if the CVI relay detects loss of vacuum or loss of CW Pumps. TBVs are latched closed on loss of vacuum (<23" Hg), and cannot be controlled remotely until the condition is cleared and the white TBV Power Reset push button is depressed on Console Center.

C. **Incorrect.** This is plausible because the TBVs will remain closed. Additionally, according to TQ-TM-104-411 (p31; Rev 5), ADV control automatically is transferred to the Control Room Back-up Loaders on loss of ICS Auto Power. Control is returned to normal following restoration of ICS Auto Power and depressing the ICS push button on Console Center. The operator may incorrectly believe that the plant conditions will result in a transfer to the Backup Loaders.

D. **Incorrect.** This is plausible because under certain conditions the system will operate in the manner. However, under the stated conditions, the TBVs will remain closed.

Technical Reference(s): TQ-TM-104-411 (p29-31; Rev 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-5 (As available)

Question Source: Bank # WTSI 46728

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to manually operate and/or monitor the steam dump valves in the control room.

The question is at the Comprehension/Analysis cognitive level because the operator must recall the setpoints of Turbine Bypass Valves and Atmospheric Dump Valves, and then given multiple points of data, conclude proper valve response.

What MUST be known:
1. The TBVs will close and be blocked from opening when Main Condenser Vacuum is low. 2. Even if Main Condenser Vacuum is restored, the TBVs will not open until the low vacuum condition is manually reset.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	073	A4.01	
	Importance Rating	3.9		

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

Proposed Question: RO Question # 20

Which ONE (1) of the following RMS monitor channels will automatically start a MAP 5 Iodine Sampler upon a Radiation Level Hi alarm?

A. RM-A-6, Auxiliary Building Vent Exhaust, gas channel.

B. RM-A-5, Condenser Vacuum Pump Exhaust, gas channel.

C. RM-A-9, Reactor Building Purge Exhaust Duct, particulate channel.

D. RM-A-8, Auxiliary and Fuel Handling Building Exhaust Duct, iodine channel.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to TQ-TM-104-661-C001 (p33-34; Rev 4), the MAP-5 remote iodine sampler stations will start automatically on a High Gas Alarm from various RMS monitors. However, According to OP-TM-MAP-C0101 (p12; Rev 1), there are no automatic actions to start a MAP-5 sampler on any alarm levels associated with RM-A-6.

B. **Correct.** According to TQ-TM-104-661-C001 (p33-34; Rev 4), the MAP-5 remote iodine sampler stations will start automatically on a High Gas Alarm from RM-A-5, 8 and 9. Additionally, according to OP-TM-MAP-C0101 (p8; Rev 1), RM-A-5/15 will start a MAP-5 Sampler on a HI Alarm.

C. **Incorrect.** This is plausible because according to TQ-TM-104-661-C001 (p33-34; Rev 4), the MAP-5 remote iodine sampler stations will start automatically on a High Gas Alarm from RM-A-5, 8 and 9, and the operator may incorrectly believe that it is the particulate channel that starts the sampler. According to TQ-TM-104-661-C001 (p23-24; Rev 4), RM-A-9 is composed of three channels, particulate monitoring, iodine monitoring and gas monitoring.

D. **Incorrect.** This is plausible because according to TQ-TM-104-661-C001 (p33-34; Rev 4), the MAP-5 remote iodine sampler stations will start automatically on a High Gas Alarm from RM-A-5, 8 and 9, and the operator may incorrectly believe that it is the iodine channel that starts the sampler. According to TQ-TM-104-661-C001 (p23-24; Rev 4), RM-A-8 is composed of three channels, particulate monitoring, iodine monitoring and gas monitoring.

Technical Reference(s): TQ-TM-104-661-C001 (p23-24, 33-34; Rev 4) OP-TM-MAP-C0101 (p8, 12; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-2 (As available)

Question Source: Bank # IR-661-GLO-2-Q02

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: None

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor the automatic operation of a radioactive effluent release in the control room. This is accomplished by identifying which, from a list of RMS channels, will automatically start a MAP-5 Iodine Sampler.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. Which RMS atmosphere monitors will automatically start a MAP-5 Sampler on a High Alarm? 2. Which specific channel, particulate, Iodine, Gas, will start the sampler?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 2.4.2	
	Importance Rating	4.5	

Main Feedwater: Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: RO Question # 21

Plant conditions:

- 50% power and stable.
- "A" and "C" Condensate Pumps are in operation.
- "A" and "C" Condensate Booster Pumps are in operation.
- The 1B Main Feedwater Pump is out of service.

Assuming no operator action occurs, which ONE (1) of the following conditions will result in an automatic reactor trip?

A. The "A" Condensate Pump trips.

B. The "C" Condensate Booster Pump trips.

C. Auxiliary Condenser Vacuum sensed at FW-LS-12A, FW-U-1A Vacuum Trip Switch, lowers to 19.5" Hg.

D. Oil pressure to the turbine bearings sensed at FW-PS-19A-1, FW-U-1A BRG Oil Low Pressure Switch, lowers to 7 psig.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** The B Condensate Pump will start in Standby. This is plausible because according to OP-TM-MAP-M0101 (p1; Rev 0), one of the causes of the automatic trip of the 1A MFWP is Loss of condensate or Condensate Booster Pumps, when needed in condensate circuit. However, according to TQ-TM-104-401 (p42; Rev 4), At least one condensate and condensate booster pump must be in operation in order to start one feedwater pump; and Two condensate and condensate booster pumps must be in

operation in order to start and operate two main feedwater pumps. Since only one MFWP is operating, the loss of the C FWBP will NOT result in an automatic trip of the 1A MFWP.

B. **Incorrect.** The B Condensate Booster Pump will start in Standby. This is plausible because according to OP-TM-MAP-M0101 (p1; Rev 0), one of the causes of the automatic trip of the 1A MFWP is Loss of condensate or Condensate Booster Pumps, when needed in condensate circuit. However, according to TQ-TM-104-401 (p42; Rev 4), At least one condensate and condensate booster pump must be in operation in order to start one feedwater pump; and Two condensate and condensate booster pumps must be in operation in order to start and operate two main feedwater pumps. Since only one MFWP is operating, the loss of the C FWBP will NOT result in an automatic trip of the 1A MFWP.

C. **Correct.**
According to OS-24, Attachment A, (p29; Rev 19), the reactor will automatically trip if Both A and B Main Feedwater pump turbines trip and > 7 % reactor power.

According to OP-TM-MAP-M0101 (p1; Rev 0), one of the causes of the automatic trip of the 1A MFWP is loss of vacuum. Since the 1B MFWP is also tripped, any condition that will trip the 1A MFWP will result in the stated conditions. According to TQ-TM-104-401 (p43-44; Rev 4), there are several conditions that will automatically trip the 1A MFWP; one of which is 9.9 inches Hg ABS rising exhaust pressure ($\leq$ 20" vacuum). FW-LS-12A: FW-U-1A VACUUM TRIP SWITCH.

IAW TQ-TM-104-401-C001, Section VI.B.3.g:
Vacuum Trip

- A vacuum trip is included in the turbine assembly, to trip out the turbine on loss of exhaust vacuum beyond a preset point.
- The trip assembly includes:
  - o A spring biased vacuum sensing bellows connected to the turbine a spring biased vacuum sensing bellows connected to the turbine exhaust.
  - o A hydraulic pressure actuated piston and its control pilot valve; a spring loaded oil dump valve and bushing (the bushing has ports in its sidewall which connect to the turbine hydraulic trip system to drain).
  - o An actuating trip finger and trip lever
  - o An air (reset) valve and interconnecting linkage.
- When the exhaust vacuum reaches the set point, the contrasting force exerted by the turbine exhaust in the bellows is overcome by the spring force, moving the pilot valve downward (through the linkage connecting lever).

D. **Incorrect.** This is plausible because according to OP-TM-MAP-M0101 (p1; Rev 0), one of the causes of the automatic trip of the 1A MFWP is low oil pressure to the turbine bearings.

IAW TQ-TM-104-401-C001, Section III.C.4.c.5).c):

- PS-19, actuates to trip the turbine in the event of low oil pressure to the turbine bearings. Contact closes at 4 psig decreasing, opens at 7 psig increasing. ECR TM 06-00143 Added 2 additional pressure switches for each feed pump [FW-PS-19A-1 (2,3) for "A" pump and FW-PS-19B-1 (2,3) for "B" pump]. A feedwater

pump trip occurs when a 2 out of 3 logic is satisfied. This MOD eliminated single failure vulnerability.

Since a 2 out of 3 logic is not given, the feedwater pump has not tripped and therefore the reactor has not tripped. Additionally, the feedwater pump will not trip unitl 4 psig, and the choice is at 7 psig.

Technical Reference(s):
OS-24, Attachment A, (p29; Rev 19)
OP-TM-MAP-M0101 (p1; Rev 0)
TQ-TM-104-401 (p43-44; Rev 4)
OP-TM-MAP-M0301 (p1; Rev 1)
OP-TM-MAP-M0103 (p2; Rev 1)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 401-GLO-5 and 10 (As available)

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

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content: 55.41 4
55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

The KA is matched because the operator must demonstrate Knowledge of system set points (6 mils vibration requires trip, but not in automatic design), interlocks (i.e. loss of one CBP will not trip only MFWP) and automatic actions (Loss of vacuum trip is active. Loss of thrust bearing wear is NOT active) associated with EOP entry conditions (i.e. the conditions have set up a situation such that the loss of the operating MFWP will result in a reactor trip).

The question is at the Comprehensive cognitive level because the operator must consider several conditions separately, and then choose the one condition that will result in an automatic

MFWP trip, which will result in an automatic reactor trip; in order to correctly answer the question.

What MUST be known:
1. What will automatically trip the 1A MFWP? 2. Do the plant conditions exist to cause an automatic MFWP trip on the loss of one Condensate Booster Pump or one Condensate Pump? 3. Has the loss vacuum trip setpoint been exceeded on the 1A Auxiliary Condenser? 4. Has the loss vacuum trip setpoint been exceeded on low oil pressure to the turbine bearings?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 2.4.21	
	Importance Rating	4.0	

Instrument Air: Emergency Procedures / Plan: 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: RO Question # 22

With the plant at 100% power, the following events occur:

- A total loss of instrument air occurs.
- The reactor has tripped.
- The crew has entered OP-TM-AOP-028, Loss of Instrument Air.
- IA Header pressure is 40 psig and slowly lowering.

The following parameters are observed:

- RCP Seal Injection flow is 15 gpm and lowering.
- ICCW flow is 575 gpm and lowering.

Which ONE (1) of the following identifies the action, if any, that should be taken with MU-V-20, Seal Injection Reactor Building Isolation Valve, IC-V-3, RB Outlet Valve, AND IC-V-4, Coolant Supply to RB?

A. No action should be taken.

B. Block open MU-V-20, ONLY.

C. Block open IC-V-3 and IC-V-4, ONLY.

D. Block open MU-V-20 and IC-V-3 and IC-V-4.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because if ICCW flow is < 550 gpm, this would be correct.

B. **Incorrect.** This is plausible because if ICCW flow is < 550 gpm, and SI flow were > 22 gpm this would be correct.

C. **Correct.** According to OP-TM-AOP-028 (p3; Rev 5), when IA header pressure is < 60 psig the operator is directed to check to see if SI flow is > 22 gpm, and if so, block open MU-V-20. Since flow is < 22 gpm, no action is taken regarding this valve. Additionally, the operator is directed to check that ICCW flow is > 550 gpm, and if so, block open IC-V-3 and 4. Since flow is > 550 gpm, these valves are blocked open. According to OP-TM-AOP-0281 (p7-8; Rev 4),Step 3.2 provides direction to direct operators to locally open MU-V-20, IC-V-3 & IC-V-4, and trip the reactor if IA pressure is less than 60 psig. Local manual control of MU-V-20 and IC-V-3 & 4 must be performed within 40 minutes of the event (IA pressure < 60 psig) in order to ensure that RCP seal cooling is not lost. However, the actions to block open these valves are contingent upon current system conditions. If seal injection flow has already been lost, then the valve is not blocked open. Similarly for ICCW, if IC flow has been lost, do not block open IC-V-3& 4. MU-V-20 or IC-V-3 & 4 may open later IAW OP-TM-AOP-041, "Loss of Seal Injection", or OP-TM-AOP-032, "Loss of Intermediate Closed Cooling".

D. **Incorrect.** This is plausible because if SI flow were > 22 gpm this would be correct.

Technical Reference(s): OP-TM-AOP-028 (p3; Rev 5)
OP-TM-AOP-0281 (p7-8; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-11 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge associated with the IA System (i.e. the action that should be taken on Loss of IA) of the parameters (SI flow, ICCW flow) and logic (> 22 gpm, > 550 gpm) used to assess the status of safety functions, such as reactor coolant system integrity. According to OP-TM-AOP-0281 (p3; Rev 4), RCS Integrity is challenged by a total loss of IA. Both methods (ICCW to thermal barrier & seal injection) of RCP seal cooling are adversely affected by loss of IA. IC-V-3, IC-V-4, and MU-V-20 fail closed on loss of air (after depletion of local air reservoir). Local operator action to block OPEN IC-V-3, 4, and MU-V-20 is needed to maintain redundant means of seal cooling.

The question is at the Comprehension/Analysis cognitive level because the operator must apply knowledge of 3 separate setpoints and evaluate greater than or less than setpoint for each, to determine which, if any, action should be taken to correctly answer the question.

What MUST be known:
1. MU-V-20 is to be blocked open if SI flow is > 22 gpm. 2. IC-V-3 and IC-V-4 are to be blocked open if ICCW flow is > 550 gpm. 3. Under the present plant conditions what actions should be taken?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	A1.03	
	Importance Rating	2.5		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Effect on instrumentation and controls of switching power supplies

Proposed Question: RO Question # 23

Plant conditions:

- 100% power.
- The Power Range Nuclear Instruments are stable and reading as follows:

NI-5	100.0
NI-6	100.5
NI-7	100.0
NI-8	99.0

- The following evolution is scheduled to take place:
    - o "A" inverter will be de-energized for maintenance IAW 1107-2B, 120 Volt Vital Electrical System.

Regarding the evolution, which ONE of the following describes:
(1) The effect on Thermal Power Heat Balance, and
(2) NI-5/NI-6 selection?

A. (1) Heat balance will remain valid throughout the entire evolution.
(2) NI-6 must be selected.

B. (1) Heat balance will remain valid throughout the entire evolution.
(2) NI-5 will remain selected.

C. (1) Heat balance will become invalid for a portion of the evolution.
(2) NI-6 must be selected.

D. (1) Heat balance will become invalid for a portion of the evolution.
(2) NI-5 will remain selected.

Proposed Answer: C

Explanation (Optional):

What must be known:

1. VBA must be divorced from “A” inverter prior to being powered from “E” inverter.
2. VBA does not automatically transfer from “A” inverter to “E” inverter, unlike ATA, which automatically transfers from “A” inverter to TRA.
3. While VBA is deenergized, heat balance will be invalid.

A. **Incorrect.**
Part 1 is incorrect: According to 1107-2B (p18; Rev 29) de-energizing VBA causes the Heat Balance calculation to be invalid. This is because PPC point A0032, Feedwater Pressure to the OTSG's, becomes invalid. Plausible if the examinee believes that the affected feedwater input is automatically removed from the calculation in this situation.

Part 2 is correct: According to 1107-2B (p13; Rev 29) Step 3.3.2.h states to select NI-6 as an input to ICS/NNI.

B. **Incorrect.**
Part 1 is incorrect: According to 1107-2B (p18; Rev 29) de-energizing VBA causes the Heat Balance calculation to be invalid. This is because PPC point A0032, Feedwater Pressure to the OTSG's, becomes invalid. Plausible if the examinee believes that the affected feedwater input is automatically removed from the calculation in this situation.

Part 2 is incorrect due to a misconception that VBA transfers on "Loss of the inverter". The transfer is only ATA which transfers to TRA.

C. **Correct.**
Part 1: According to 1107-2B (p18; Rev 29) de-energizing VBA causes the Heat Balance calculation to be invalid. This is because PPC point A0032, Feedwater Pressure to the OTSG's, becomes invalid.

Part 2: According to 1107-2B (p13; Rev 29) Step 3.3.2.h states to select NI-6 as an input to ICS/NNI.

D. **Incorrect.**
Part 1 is correct: According to 1107-2B (p18; Rev 29) de-energizing VBA causes the Heat Balance calculation to be invalid. This is because PPC point A0032, Feedwater Pressure to the OTSG's, becomes invalid.

Part 2 is incorrect due to a misconception that VBA transfers on "Loss of the inverter". The transfer is only ATA which transfers to TRA.

Technical Reference(s): 1107-2B (pgs 13, 18; Rev 29) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP-DBIG PCO-3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including the effect on instrumentation and controls of switching power supplies. This is accomplished by presenting a set of plant conditions in which VBA will be de-energized requiring the operator to assess the impact, and choose the correct course of action.

The question is at the Memory cognitive level because the operator must recall two bits of information, to correctly answer the question.

What MUST be known:
1. How does de-energizing VBA affect the Heat Balance? 2. What is the alternative method for controlling power when the heat balance is invalid?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A2.03	
	Importance Rating	3	

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: High/low CCW temperature

Proposed Question: RO Question # 24

Plant conditions:

- 100% power.
- Intermediate Closed Coolers are in service as follows:
  - IC-C-1A is in full cooling operation.
  - IC-C-1B is throttled to 50% cooling.
- IC Cooler Outlet temperature IC-TI-6 indicates 115°F.

Which ONE (1) of the following identifies the concern with this condition, AND the action that must be taken?

A. Potential overheating of the CRDMs;
Throttle OPEN IC-C-1B River Inlet Valve NR-V-10B.

B. Potential overheating of the CRDMs;
Throttle OPEN IC-C-1B River Outlet Valve NR-V-15B.

C. Potential condensation on the CRD cooling coils;
Throttle CLOSED IC-C-1B River Inlet Valve NR-V-10B.

D. Potential condensation on the CRD cooling coils;
Throttle CLOSED IC-C-1B River Outlet Valve NR-V-15B.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the temperature control is accomplished by throttling the inlet

valve rather than the outlet valve.

B. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-541-000 (p4; Rev 12) to minimize the possibility of forming condensate in the CRD stator water jacket and provide adequate CRD stator cooling, IC outlet temperature should be maintained between 90°F and 100°F on IC-6TI (CR). According to TQ-TM-104-531 (p45; Rev 6), the low limit of 90°F is to prevent condensation on the CRD, cooling coils and the upper limit of 100°F is to prevent overheating the CRDMs. Since the temperature is too high, the concern is the overheating of the CRDMs. According to OP-TM-541-000 (p21; Rev 12), in the Operating Mode both IC-C-1A River Outlet Valve NR-V-15A and IC-C-1B River Outlet Valve NR-V-15B are throttled IAW OP-TM-541-461. According to OP-TM-541-461 (p3; Rev 6) the operator is directed to THROTTLE NR-V-15A to maintain IC cooler outlet temperature IC-6TI (CR) between 90°F and 100°F. If NR-V-15A is full OPEN, which is the case in this situation, then THROTTLE NR-V-15B to maintain IC cooler outlet temperature IC-6TI (CR) between 90°F and 100°F.

C. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that the IC System temperature is too low, and if this were true, then condensation on the CRDM cooling coils would be the concern.

Technical Reference(s): OP-TM-MAP-C0302 (p1; Rev 3)
TQ-TM-104-531-C001 (p36, Rev 6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate ability to (a) predict the impacts of high temperature condition in the IC System (Overheat the CRDMs), and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences (i.e. choose the preferred action between two actions listed in the procedure).

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate an understanding of the impact of a hi/low temperature condition to correctly answer the question.

What MUST be known:
1. What is the normal temperature of the IC Cooler Outlet? 2. What is the concern if IC Cooler Outlet Temperature is too high? 3. What is the concern if IC Cooler Outlet Temperature is too low? 4. If the IC Cooler Outlet Temperature is too high, what action is required?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A3.02	
	Importance Rating	3.1		

Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question: RO Question # 25

Plant conditions:

- Plant startup in progress.
- Reactor power is 10%, with the Main Turbine accelerating to 1800 rpm.
- Main Steam Isolation Valves, MS-V-1A, B, C and D are all open.
- All Turbine Bypass Valves (TBVs) in automatic control.

Event:

Time	Action
0 seconds	MS-V-3D, E, and F automatic demand signal fails to 100 percent.
25 seconds	The RO transfers MS-V-3D, E, and F control to Hand, and THEN RO throttles MS-V-3D, E, and F to terminate steam pressure reduction.

Pressure Indication	T=0	T=45 seconds
OTSG 1A Pressure	908	825
"A" Steam line SP10A-PT-2	906	895
"B" Steam line SP10A-PT-1	906	895
OTSG 1B Pressure	906	896
"C" Steam line SP10B-PT-1	904	894
"D" Steam line SP10B-PT-2	904	894

Based on these conditions, which ONE (1) of the following describes the mechanical condition of the Main Steam Isolation Valves, MS-V-1A, B, C and D?

A. MS-V-1A, B, C and D are seated.

B. MS-V-1A and MS-V-1B are seated;
MS-V-1C and MS-V-1D held open with steam pressure.

C. MS-V-1C and MS-V-1D are seated;
MS-V-1A and MS-V-1B held open with steam pressure.

D. MS-V-1A, B, C and D are held open with steam pressure.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because pressure is reduced in BOTH OTSGs over the 45 seconds of the event, and the operator may incorrectly believe that all MSIVs are now seated.

B. **Correct.** According to TQ-TM-104-411-C001 (p4; Rev 5), there are two steam lines from each steam generator to the turbine for a total of four lines. The only cross connection between leads is in the turbine steam chest between the turbine stop valves and control valves. According to TQ-TM-104-411-C001 (p13-14; Rev 5) MS-V-1A/B/C/D are Motor Operated Angle Stop Check Valves. If the Main Turbine Stop Valves fail to close for a rupture upstream of the MSIV (i.e., inside containment), automatic closure of the MSIV check valves will prevent flow from the unaffected OTSG out the break (reverse flow from the turbine valve chest). When the event occurs, a 70 psid reverse differential pressure is created from the Turbine Chest to the A OTSG. This will seat MS-V-1A and B. On the other hand, the differential pressure between the Turbine Chest to the B OTSG is not in the reverse direction, and MS-V-1C and D will be held open by steam pressure.

C. **Incorrect.** MS-V-1C and MS-V-1D are on OTSG 1B steam lines. This is plausible because TBV numbering is reversed between OTSG 1A and 1B (MS-V-3A-C are on OTSG 1B, while MS-V-3D-F are on OTSG 1A); and the operator may incorrectly believe that MS-V-1C and D are on the 1A OTSG, rather than the 1B OTSG. Additional plausibility is merited because OTSG 1B pressure has been reduced due to reduction in Tave induced by opening OTSG 1A TBVs.

D. **Incorrect.** This is plausible because this is the normal condition because this was the condition of the MSIVs at the beginning of the event, and the operator may incorrectly believe that their status has not changed.

Technical Reference(s): TQ-TM-104-411-C001 (p4, 13; Rev 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-3 (As available)

Question Source:	Bank #	IR-411-GLO-3-Q03	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	None

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

10 CFR Part 55 Content:	55.41	4
	55.43	

Secondary coolant and auxiliary systems that affect the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the MRSS, including isolation of the MRSS. This is accomplished by requiring the operator to evaluate plant conditions and identify the effect of the MSIVs that the given event will have.

The question is at the Comprehension/Analysis cognitive level because the operator must know design features of the MSIV (i.e. stop check), and then evaluate plant conditions and identify the status of the MSIVs to correctly answer the question.

What MUST be known:
1. The MSIVs are stop check valves and will close on a reverse differential pressure. 2. MS-V-1A and MS-V-1B are connected to the 1A OTSG. 3. MS-V-1C and MS-V-1D are connected to the 1B OTSG. 4. The 1A and 1B OTSG are cross-connected at the Turbine chest.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	006	K3.01	
	Importance Rating	4.1		

Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: RCS

Proposed Question: RO Question # 26

Plant conditions:

- "C" steam line has ruptured in containment.
- 4 psig ESAS occurred.
- Makeup pump suction from BWST, MU-V-14A and B have failed to open.
- Intermediate Closed Cooling Pumps, IC-P-1A and IC-P-1B have tripped.

Currently:

- "B" OTSG is isolated.
- Cooldown is halted.
- RCS pressure is 1150 psig.
- RCS temperature is 530°F.
- Makeup tank level is 16 inches and lowering.

Which ONE (1) of the following actions is required?

A. Ensure all four Reactor Coolant Pumps are tripped, ONLY.

B. Ensure all four Reactor Coolant Pumps are tripped AND open DH-V-7A and DH-V-7B.

C. Place ES selected HPI pumps (MU-P-1A and MU-P-1C) in Pull-To-Lock (PTL), ONLY.

D. Place all HPI pumps in Pull-To-Lock (PTL) AND ensure all 4 Reactor Coolant Pumps are tripped.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** Plausible for a loss of SCM incorrect SCM is approximately 28°F.

B. **Incorrect.**
Plausible for a loss of SCM and DH-V-7A/B would provide suction for MU-P-1A,B,C, however incorrect SCM is approximately 28°F, and procedure directs the tripping of MU-P's if MU tank level went below 18 inches.

C. **Incorrect.** OP-TM-211-901 4.2.1.1 IF At Any Time MU tank < 18" PLACE all the following in PTL MU-P-1A, MU-P-1B, MU-P-1C. This is only partially correct but plausible if the examinee believes that enough Makeup inventory exists to run the normally-running Makeup Pump.

D. **Correct.** OP-TM-211-901 4.2.1.1 If At Any Time MU tank < 18" PLACE all the following in PTL MU-P-1A, MU-P-1B, MU-P-1C. Additionally, With a loss of Makeup and Intermediate Closed Pumps, RCP's should trip on interlock. The operator is required to ensure this action occurs and manually perform the action if it does not happen automatically.

Technical Reference(s):	OP-TM-211-901 Rev 6 step 4.2.1.1 TQ-TM-104-211-C001 (p47; Rev5)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-901-PCO-5 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	8
	55.43	

Components, capacity, and functions of emergency systems.

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect that a loss

or malfunction of the ECCS will have on the RCS. This is accomplished by presenting the operator with a set of conditions where the candidate must recognize the need to secure Makeup to the RCS. Therefore the ECCS malfunction results in a loss of inventory makeup. Additionally, RCP's will be secured to avoid overheating the RCP seals and therefore avoiding an unisolable RCS rupture.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the plant conditions to determine procedural actions required to mitigate conditions given.

What MUST be known:
1. Failure of both BWST suction flow path AND less than 18 inches in the MU tank requires placing all Makeup pumps in PTL. 2. SCM is greater than 25°F. 3. A loss of Seal Injection (from Makeup) and ICCW will lead to RCP seal overcooling.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064	K4.02
	Importance Rating	3.9	

Knowledge of ED/G system design feature(s) and/or inter-lock(s) which provide for the following: Trips for ED/G while operating (normal or emergency)

Proposed Question: RO Question # 27

Plant conditions:

- 100% power with ICS in full automatic.
- The 1600 psig ESAS inadvertently actuates.

Which ONE (1) of the following identifies ~~a set of~~ An Automatic Diesel Shutdown~~s~~ that is bypassed under these conditions?

4/24/12

A. Engine Overspeed.

B. 86G Generator Lockout.

C. High Crankcase Pressure.

D. Low Lube Oil Pressure (Running).

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to TQ-TM-104-861-C001 (p71; Rev 8), the EDG can be automatically shutdown by Engine Overspeed; and the operator may incorrectly believe that this is bypassed under ES conditions. Plausibility is strengthened by the fact that this trip is unique in that it is purely mechanical; and by the fact that both the OPL1, 2 and 3 and High Crankcase Pressure trips utilize the SDR to create the trip.

B. **Incorrect.** This is plausible because according to TQ-TM-104-861-C001 (p71; Rev 8), the EDG can be automatically shutdown by the 86G Generator Lockout; and the operator may incorrectly believe that this is bypassed under ES conditions. Plausibility is strengthened by the fact that it is the only trip that is operated via the 5/5A Relays;

and by the fact that both the OPL1, 2 and 3 and High Crankcase Pressure trips utilize the SDR to create the trip.

C. **Correct.** According to TQ-TM-104-861-C001 (p71; Rev 8), the EDG can be automatically shutdown by several protective trips. These are (1) Engine Overspeed, (2) Low Lube Oil Pressure, (3) High Crankcase Pressure, and the Generator 86 Lockout. According to TQ-TM-104-861-C001 (p55; Rev 8), the High Crankcase Pressure trip is bypassed under ES conditions.

D. **Incorrect.** This is plausible because according to TQ-TM-104-861-C001 (p71; Rev 8), the EDG can be automatically shutdown by Low Lube Oil Pressure. Additionally, according to TQ-TM-104-861-C001 (p18; Rev 8), the EDG has two protective Low Lube Oil Automatic Shutdowns; (1) OPLS and (2) OPL1, 2 and 3. The OPLS is for prolonged low speed, no load operation, and is activated at 20 sec after 250 rpm. It is set at 7 psi decreasing pressure and triggers both alarm and shutdown, however, this trip is defeated by ES or if governor is set at 900 RPM. The operator may incorrectly believe that the trip initiated by OPL1, 2 and 3 is similar.

Technical Reference(s): TQ-TM-104-861-C001 (p18, 55 and 71; Rev 8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-12 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the operator must demonstrate knowledge of the EDG system interlock(s) which provide for automatic trips of ED/G during emergency operation.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. When the EDG is operating under ES conditions, what conditions will trip the engine? 2. When the EDG is operating under ES conditions, what protective devices will NOT trip the engine?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	2.1.7	
	Importance Rating	4.4		

Residual Heat Removal: Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: RO Question # 28

Sequence of events:

- Reactor trip due to low RCS pressure.
- Automatic ES actuations:
  - 1600 psig.
  - 500 psig.
  - 4 psig RB pressure.
  - 30 psig RB pressure.

Current plant conditions:

- RCS pressure is 270 psig and stable.
- RB flood level is 32 inches and rising.
- RB pressure is 7 psig and lowering.
- BWST level is 14.5 feet and lowering.

Based on the conditions above, which ONE of the following describes:
(1) The operational implications related to continuing to lower BWST and raising RB flood levels, and
(2) The required action?

A. (1) Vital instrumentation could be damaged by flooding in the RB;
(2) Secure RB Spray pumps.

B. (1) Vital instrumentation could be damaged by flooding in the RB;
(2) Transfer LPI pump suction to RB sump via RB Sump Recirculation.

C. (1) Adequate pump NPSH (Net Positive Suction Head) will be lost;
(2) Secure LPI pumps.

D. (1) Adequate pump NPSH (Net Positive Suction Head) will be lost;
(2) Transfer HPI Pump suction to LPI pump discharge via "Piggyback" Mode.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-EOP-0101 (p62; Rev 3), Action is also initiated if RB Flood level is greater than 54". This level can only be reached in an event where significant additional injection beyond the normal BWST volume has been injected. Continued injection from the BWST would flood some RB instruments. The operator may incorrectly believe that RB flooding, is the most significant operational implication. If so, it is logical to think that the RB Spray System must be shutdown.

B. **Incorrect.** This is plausible because according to OP-TM-EOP-0101 (p62; Rev 3), Action is also initiated if RB Flood level is greater than 54". This level can only be reached in an event where significant additional injection beyond the normal BWST volume has been injected. Continued injection from the BWST would flood some RB instruments. The operator may incorrectly believe that RB flooding, is the most significant operational implication. If so, it is logical to think that the LPI pump suction must be aligned for recirculation to limit the flow of water into the RB.

C. **Incorrect.** This is plausible because the LPI Pumps are a large draw on the RWST, and level is rapidly approaching the point where the LPI pumps will cavitate. However, the LPI Pumps are also providing the source of core cooling at the present conditions.

D. **Correct.** According to OP-TM-EOP-0101 (p59 and 61; Rev 3), the purpose of Guide 20 is to align Auxiliary Building equipment prior to ECCS suction transfer to Containment sump. The HPI system NPSH requirements cannot be satisfied during the ECCS suction transfer sequence to the Containment sump due to gravity flow from the BWST to the sump lowering the available head. For this reason, if HPI termination criteria are satisfied, the system is shutdown, or if HPI cannot be terminated, then the HPI suction is transferred to the LPI discharge and permitted to operate in accordance with Rule 2, "HPI / LPI Throttling". This is the case here. According to OP-TM-EOP-0101 (p62; Rev 3), The purpose of Guide 21 is to transfer suction of Decay Heat and Building Spray pumps from the BWST to the RB sump. This transition must be initiated if BWST level is < 15 feet or RB flood level is > 54 inches. In this case, the BWST level is low. Action is initiated at BWST < 15 ft to ensure the actions are completed in a timely manner. The transition from the BWST to the RB sump may be time critical evolution. Operators should be well prepared to execute this sequence after BWST level reaches 15 ft. In a LBLOCA scenario if all ECCS functions operate as designed and ambient temperature is cold, flow from the BWST will be high and will increase when DH-V-6A & B are open. In this limiting case, action within seconds of reaching setpoint is required to complete the transition before air could be drawn into the DH pump suction due to vortex formation in the BWST. In the limiting scenario, BWST level could be dropping at 2 ft / minute during the transition.

Technical Reference(s):	OP-TM-EOP-0101 (p59 and 61-62; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	212-GLO-12	(As available)

Question Source:	Bank #	IR-212-GLO-12-Q02	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2003

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. This is accomplished by presenting the operator with a set of plant conditions characteristic of a medium sized LOCA, with lowering BWST Levels, and rising flood waters in the RB, and then requiring the operator to choose the most significant operational implication, and take action based on this.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then apply this information to the presented situation, in order to correctly answer the question.

What MUST be known:
1. What is the level at which the BWST starts to become sufficiently low such that the LPI Pumps, and others, taking suction from the tank are threatened? 2. What is the flooding level in the RB at which vital instruments will be adversely impacted? 3. Based on the conditions presented, what actions should be taken?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	002	A1.03	
	Importance Rating	3.7		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Temperature

Proposed Question: RO Question # 29

Plant conditions:

- A plant cooldown is in progress in accordance with 1102-11, Plant Cooldown.
- Current reactor coolant system temperature is 250°F.

Which ONE (1) of the following describes the maximum RCS cooldown rate limit for the conditions, AND the required time interval (per 1102-11, Plant Cooldown) for recording data during the cooldown?

A. 30˚F/hour; AND
Every 5 minutes.

B. 30˚F/hour; AND
Every 30 minutes.

C. 100˚F/hour; AND
Every 5 minutes.

D. 100˚F/hour; AND
Every 30 minutes.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because it would be correct if the plant computer were unavailable.

B. **Correct.** 1st part correct, 2nd part correct. According to 1102-11 (p4; Rev 140), the maximum allowable RCS cooldown rate when RCS temperature is less than or equal to 255°F, is less than 30°F/hour or 15°F/30 minutes. According to 1102-11, Enclosure 4

(5of7; Rev 140), every 30 minutes, the operator is directed to plot a point and record the time on the P/T curves (Encl 4 Fig 1 or 1A). If the plant computer calculated cooldown rates are unavailable, the operator is instead directed to complete Enclosure 4 Data Sheet every 5 minutes. A note prior to this step identifies that these requirements are for compliance with Tech. Spec. 3.1.2. Since there is no indication that the plant computer is unavailable, 30 minute intervals are correct.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because this would be correct if RCS temperature were > 255°F, and the plant computer were unavailable.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because this would be correct if RCS temperature were > 255°F.

Technical Reference(s): 1102-11 (p4; Rev 140)
1102-11, Enclosure 4 (5of7; Rev 140) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-010-PCO-2 (As available)

Question Source: Bank #

Modified Bank # IR-GOP-010-PCO-2-Q01 (Note changes or attach parent)

New

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor changes in RCS Temperature to prevent exceeding Technical Specification limits. This is accomplished by requiring the operator to identify the maximum cooldown rate for a given plant condition,

and the time intervals for monitoring during cooldown.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. Maximum cooldown rates for various plant temperatures. 2. The time interval for monitoring the RCS temperature during plant cooldown, under normal circumstances (i.e. Plant computer is available).

The question is significantly modified from IR-GOP-010-PCO-2-Q01 because it includes another element of the question (monitoring time interval). The question is significantly modified from Q62 on NRC Exam 2008-2 because the conditions in the stem have been changed such that one of the three distractors in the original question becomes the correct. According to NUREG-1021, ES-701, Page 3 of 18, to be considered a significantly modified question, at least one pertinent condition in the stem and at least one distractor must be changed from the original bank question. Changing the conditions in the stem such that one of the three distractors in the original question becomes the correct answer would also be considered a significant modification.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	014	K1.02	
	Importance Rating	3.0		

Knowledge of the physical connections and/or cause-effect relationships between the RPIS and the following systems: NIS

Proposed Question: RO Question # 30

Plant conditions:

- A transient reduced power from 100% to 85%.
- When power stabilized MAP G-2-1, CRD PATTERN ASYMMETRIC, alarmed.
- One Group 7 Rod was 8 inches higher than the Group 7 Absolute Group Average Position.
- The crew suspected a position indication problem and entered OP-TM-622-416, Evaluating PI Problems.

In accordance with OP-TM-622-416, which ONE (1) of the following identifies parameters that the operator must observe to distinguish between an actual misaligned rod and a position indication problem?

A. Flux imbalance meters and FIDMS.

B. Flux imbalance meters and Core Exit Thermocouples.

C. Power Range Recorder (CC) and FIDMS.

D. Power Range Recorder (CC) and Core Exit Thermocouples.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-MAP-G0201 (p1; Rev 2a), the CRD Pattern Asymmetric will alarm when any rod "7" (5 percent) from its Absolute Group Average Position. According to Step 4.1, if an RPI problem is suspected, the operator is directed to evaluate asymmetric rod condition IAW OP-TM-622-416, Evaluating PI Problems. According to OP-TM-622-416 (p2; Rev 3), among other RPI System indications the operator must observe the imbalance meters on CC

for flux imbalance, and the FIDMs printouts.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that OP-TM-622-416 directs the operator to consider the CETs when analyzing for a stuck rod.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that OP-TM-622-416 directs the operator to consider the Power Range Recorder when analyzing for a stuck rod.

D. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

Technical Reference(s): OP-TM-MAP-G0201 (p1; Rev 2a)
OP-TM-622-416 (p2; Rev 3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO-10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the operator must demonstrate knowledge of the cause-effect relationships between the RPIS and the NIS. If the rod is actually misaligned as indicated by RPI, the stated parameters will confirm this. On the other hand, if the failure has occurred in the RPI System, the stated parameters will also confirm this as well.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. Which parameters does OP-TM-622-416 direct the operator to observe when evaluating PI problems.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	041	2.1.20	
	Importance Rating	4.6		

Steam Dump/Turbine Bypass Control: Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 31

Plant conditions:

- A Tube Rupture has occurred on OTSG "B".
- The crew is performing actions of OP-TM-EOP-005, OTSG Tube Leakage.
- Reactor power is 10%.
- The crew is preparing to trip the reactor.

Which ONE (1) of the following correctly completes the statement below?

Prior to tripping the reactor the operator should ____(1)____, and just after the trip it is expected that the TBVs will move, or be moved, in the ____(2)____ direction initially.

A. (1) VERIFY the TBVs in AUTO
(2) OPEN

B. (1) VERIFY the TBVs in AUTO
(2) CLOSED

C. (1) PLACE the TBVs in HAND
(2) OPEN

D. (1) PLACE the TBVs in HAND
(2) CLOSED

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the TBVs are

normally in AUTO, and the operator may incorrectly believe that they should be left in AUTO in the present situation.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may misunderstand how the valve operates, and/or how it controls OTSG pressure, and incorrectly believe that the valve will need to go open initially.

D. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-005 (p13; Rev 7) Step 22 and 23, the operator is directed to PLACE TBVs to HAND, TRIP the reactor, and THROTTLE TBVs to achieve the following: (1) Prevent MSSV lift, and (2) Stabilize RCS temperature. According to OP-TM-EOP-0051 (p11; Rev 2), prior to tripping the reactor, with reactor power at slightly less than 10%, the TBVs are placed in hand. At this TBV position OTSG pressure will decrease when the reactor is tripped. This means that in order to stabilize OTSG pressure, and ultimately RCS temperature, the TBVs will need to be initially closed.

Technical Reference(s): OP-TM-EOP-005 (p13; Rev 7)
OP-TM-EOP-0051 (p11; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP005-PCO-1 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate ability to interpret and execute

procedure steps. This is done by presenting the operator with a set of plant conditions that will require the implementation of a procedure step that provides specific direction on how the TBVs must be controlled. The operator will be required to identify the control mechanism (i.e. AUTO or HAND), and then predict the direction the valves will need to be moved initially.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then apply that information, demonstrating an understanding of how RCS temperature can be controlled by controlling OTSG pressure in a post trip setting; to correctly answer the question.

What MUST be known:
1. What mode of operation does EOP-005 require the TBVs to be in just prior to manually tripping the reactor at 10%? 2. What will happen to OTSG pressure on the reactor trip, with the TBVs in HAND, but opened to where they were in AUTO, just prior to their being placed in HAND? 3. How does the operator stabilize RCS temperature post -trip in the given situation?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	075	K2.03
	Importance Rating	2.6	

Knowledge of bus power supplies to the following: Emergency/essential SWS pumps

Proposed Question: RO Question # 32

Plant conditions:

- 100% power
- Nuclear River Pumps NR-P-1B and NR-P-1C are ES selected.
- NR-P-1A and NR-P-1B are operating.
- NR-V-4A & B are OPEN for Circulating Water Flume Makeup.

Event:

- A 4 psig RB Pressure ESAS occurs.
- The supply breaker to 480V Bus 1R trips open, and cannot be re-closed.

Which ONE (1) of the following identifies how the Nuclear River Water Pumps and Discharge Valves respond to these events?

A. Discharge valves on all three pumps are open;
NR-P-1C is operating ONLY.

B. Discharge valves NR-V-1B and NR-V-1C are open ONLY;
NR-P-1B and NR-P-1C are operating.

C. Discharge valves on all three pumps are open;
NR-P-1B and NR-P-1C are operating.

D. Discharge valves NR-V-1B and NR-V-1C are open ONLY;
NR-P-1C is operating ONLY.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to TQ-TM-104-531-001 (p25; Rev6) NR-P-1A is powered from the 1R 480V ES Bus and NR-P-1C is powered from the 1T 480V ES Bus. On the other hand, NR-P-1B receives power from either the 1R or the 1T Safeguards Buses via Kirk Key Interlocked breakers. Also, according to TQ-TM-104-531-001 (p37; Rev6), the Non-ES Selected NR pump will trip on an ES actuation. With NR-P-1B and NR-P-1C ES Selected NR-P-1B must be racked in on the 1R 480 volt bus. With NR-P-1A and NR-P-1B initially operating their discharge valves would be open, they remain open on pump shutdown. NR-P-1C would receive two start signals one from 4# ES and another from pump disagreement (tripped pump), and its discharge valve would open on pump start.

B. **Incorrect.** This is plausible for the misconception that NR-P-1B would be powered from its normal 1T 480 volt bus, and that NR-V-1A would close on pump shut down (as a DR-P-1 discharge would).

C. **Incorrect.** This is plausible for the misconception that NR-P-1B would be powered from its normal 1T 480 volt bus, and the correct belief that NR-V-1A would remain open.

D. **Incorrect.** This is plausible for the knowledge that only the "C" pump has power, but the misconception that the "A" discharge valve being non-ES would have closed, with the ES pumps discharge valves remaining open.

Technical Reference(s): TQ-TM-104-531-001 (p25 and 37; Rev6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 533-GLO-4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of bus power supplies to the Emergency/essential SWS pumps. IN this case the ESW Pumps were considered the NR Pumps. The knowledge is demonstrated by presenting a set of plant conditions to the operator in which an ESAS is present , and an electrical fault occurs on one train, unrelated to the DR Pumps and Valves. The operator is required to evaluate the impact, and in the process of doing must demonstrate knowledge of the Bus power supplies.

The question is at the Comprehension/Analysis cognitive level because the operator must integrate the knowledge of ES selection separation of trains to deduce that the "B" pump is powered from the "A" train, and then must recognize the loss of "A" train with the loss of 1R 480 volt bus, further understanding and relating the interlocking of discharge valves with pumps.

What MUST be known:
1. NR-P-1B is powered from 1R 480 volt bus when NR-P-1B and "C" are ES selected. 2. NR-P-1A is powered from 1R 480 volt bus. 3. Running pump discharge valves are open and get no close signal on pump shutdown. 4. 3rd pump will start and is interlocked to open its discharge valve.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	071	A2.09	
	Importance Rating	3.0		

Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck-open relief valve

Proposed Question: RO Question # 33

Plant conditions:

- During a Refueling Outage Waste Gas Delay Tank Relief Valve WDG-V-12 was improperly set at 2.2 psig.
- Shortly after startup the relief valve lifted and stuck in the open position.

Which ONE (1) of the following identifies the Radiation Monitor that would be able to detect this condition, AND the required action in accordance with 1104-27, Waste Disposal Gaseous?

A. Auxiliary Building Vent Exhaust Monitor RM-A-6;
Gag the relief valve closed and continue Waste Gas Delay Tank operations.

B. Auxiliary Building Vent Exhaust Monitor RM-A-6;
Bypass and remove the Waste Gas Delay Tank from Service.

C. Auxiliary and Fuel Handling Building Exhaust Duct Monitor RM-A-8;
Gag the relief valve closed and continue Waste Gas Delay Tank operations.

D. Auxiliary and Fuel Handling Building Exhaust Duct Monitor RM-A-8;
Bypass and remove the Waste Gas Delay Tank from Service.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that the relief valve lifts to the local atmosphere, and if so, would ultimately get detected by RM-A-6 before RM-A-8.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because relief valves that have failed can typically be gagged closed. However, the procedure for Relief Valves, 1410-V-3, Relief Valve Maintenance, Step 5.3 states:
It is not the intent of this procedure to correct active system leakage at pressure.

D. **Correct.** 1st part correct, 2nd part correct. According to Drawing 302-694 (Rev 45), Relief Valve WDG-V-12 sits atop the Waste Gas Delay Tank. It has had a modification that raised the setpoint from 2.2 psig to 8 psig because there were too many unplanned releases occurring. If the valve lifts and sticks open, according to Drawing 302-841 (Rev 50), it would flow into the Plant Vent and be detected by RM-A-8. This is substantiated by OP-TM-MAP-C0101 (p14; Rev 1), which cites Hi gas release activity from W.G. Decay Tks as one reason that the monitor would enter the alarm condition. The Relief Valves from the WGDT relieve into the same header as the relief on the Waste Gas Delay Tank. This Alarm response procedure requires that the operator isolate the source of radioactive release if possible. According to 1104-27 (p16; Rev 82), detailed instructions are provided for bypass and isolating the Waste Gas Delay Tank while at power, which will isolate the leak.

Technical Reference(s): Drawing 302-694 (Rev 45)
Drawing 302-841 (Rev 50)
OP-TM-MAP-C0101 (p14; Rev 1) (Attach if not previously provided)
1104-27 (p16; Rev 82)

Proposed References to be provided to applicants during examination: None

Learning Objective: 231-GLO-10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the operator must demonstrate the ability to (a) predict the impacts of a Waste Gas Disposal System stuck open relief valve ; and (b) based on those predictions, use procedures to correct the situation.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then apply system knowledge regarding specific action to take, to correctly answer the question.

What MUST be known:
1. What RMS will detect a lift WGS Relief Valve? 2. What does MAP-C-101 direct the operator to do regarding the Hi Rad Alarm on RM-A-8? 3. What actions will 1104-27 permit to be taken on the Waste Gas Delay Tank while at power?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029 K4.03	
	Importance Rating	3.2	

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Automatic purge isolation

Proposed Question: RO Question # 34

Plant Conditions:

- Reactor Shutdown in progress.
- RB Purge is in progress in preparation for the start of a Refueling Outage.

Event Occurrence:

- An inadvertent actuation of the "A" Train 4# RB Pressure has occurred.

Which ONE (1) of the following identifies the RB equipment realignment and RB purge status?

A. Purge Exhaust valve AH-V-1B closes;
Purge Supply valve AH-V-1D closes;
RB Purge Exhaust Fan, AH-E-7A, automatically trips.

B. Purge Exhaust valve AH-V-1B closes;
Purge Supply valve AH-V-1D closes;
RB Purge Exhaust Fans, AH-E-7A and AH-E-7B, continue to run.

C. Purge Exhaust valves AH-V-1A and 1B close;
Purge Supply valves AH-V-1C and 1D close;
RB Purge Exhaust Fan, AH-E-7A, automatically trips.

D. Purge Exhaust valves AH-V-1A and 1B close;
Purge Supply valves AH-V-1C and 1D close;
RB Purge Exhaust Fans, AH-E-7A and AH-E-7B, continue to run.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part (Valves) correct, 2nd part (Exhaust Fan status) wrong. This is plausible because the operator may incorrectly believe that the ESAS signal, or the valve position changes will result in the Exhaust Fan tripping.

B. **Correct.** 1st part (Valves) correct, 2nd part (Exhaust Fan status) correct. According to 1105-3, Attachment 1 (p14; Rev 51), both AH-V-1B and AH-V-1D will close on the Train A 4 psig RB Pressure Actuation. Each Train closes one inside and one outside supply and exhaust valve. According to TQ-TM-104-824-C001 (p37-38; Rev 4) the RB Purge Supply and Exhaust Fans (AH-E-6 and 7 A&B) can run with no Purge Valves OPEN. Both sets of Fans will trip on AH-F-1 Fire Detection. Additionally, AH-E-6 trips on high duct temperature and AH-E-7 trip. Therefore, under the stated conditions, there is no signal that will automatically trip the Purge Supply and Exhaust fans.

C. **Incorrect.** 1st part (Valves) wrong, 2nd part (Exhaust Fan status) wrong. See A and D.

D. **Incorrect.** 1st part (Valves) wrong, 2nd part (Exhaust Fan status) correct. This is plausible because the operator may incorrectly believe that one train of ESAS closes ALL the RB Purge isolation valves, as is the case if RM-A-9 were to alarm.

Technical Reference(s):	1105-3, Attachment 1 (p14; Rev 51) TQ-TM-104-824-C001 (p37-38; Rev 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	824-GLO-2 and 8	(As available)

Question Source:	Bank #	QR-824-GLO-10-Q02	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2010

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

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Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate Knowledge of design feature(s) and/or interlock(s) (i.e. how does system respond to one train of ESAS actuating, and how does this affect Exhaust fan operation) which provide for Automatic purge isolation.

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate plant conditions and determine how the RB Purge System isolation valves have responded to these conditions, and then based on these conditions determine how the system exhaust fans will be affected.

What MUST be known:
1. How does a Train A 4# actuation affect the Purge Supply and Exhaust Valves? 2. Can the AH-E-7 Fans operate with the Purge Supply and Exhaust Valves closed? 3. What will cause the AH-E-7A Fan to trip? 4. What will cause the AH-E-7B Fan to trip?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072	K3.02
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Fuel handling operations

Proposed Question: RO Question # 35

Given:

1. RM-A-4, Fuel Handling Bldg. Exhaust
2. RM-A-8, Aux. and Fuel Handling Building Exhaust Duct
3. RM-A-14, ESF Ventilation
4. RM-G-9, Fuel Handling Area FHB

Which of the above listed Radiation Monitors have both:

- An interlock to trip AH-E-10, FHB Supply Fan, **AND**
- An interlock to close AH-D-120, 121 and 122, FHB Isolation Dampers?

A. 1 and 3.

B. 1 and 4.

C. 2 and 3.

D. 2 and 4.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to TQ-TM-104-829-C001 (p5; Rev 2) the ESF Ventilation System is used to continuously monitor exhaust air from the fuel handling area for radioactivity while handling irradiated fuel. The operator may incorrectly believe that it has automatic actions associated with it.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-661-001 (p16; Rev 4) RM-G-9 has an interlock provided to stop the Fuel Handling Building air supply fan AH-E-10 and close dampers, which helps to promote a negative pressure in the Fuel Handling Building. Also according to TQ-TM-104-661-001 (p66; Rev 4), the Fuel Handling Building air intake and exhaust are controlled by dampers that close on a signal from radiation monitor RM-G-9, or radiation monitor RM-A-4 Gas (with local and control room alarms) and assures separation of the Auxiliary Building Ventilation and Fuel Handling Building Ventilation Systems. This same information is identified in TQ-TM-104-829-001 (p11-12; Rev 2) and in MAP C-1-1 (p5 and 26; Rev 1).

C. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because RM-A-8 is a radiation monitor associated with the FHB, and the operator may incorrectly believe that it isolates the FHB Vent System upon detecting High Radiation. Additionally, according to MAP C-1-1 (p16; Rev 1), this detector has automatic functions, including the automatic tripping of AH-E-10. However, it does NOT also close the FHB Isolation Dampers.

Technical Reference(s): TQ-TM-104-661-001 (p16 and 66; Rev 4)
TM-104-829-001 (p5, 11-12; Rev 2)
MAP C-1-1 (p5, 16, and 26; Rev 1)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-15 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	11
	55.43	

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect that a loss of RM-G-9 or RM-A-4 will have on the FHB Ventilation System, which in-turn will affect the ability to continue Fuel Handling Operations. According to 1505-1 Data Sheet 3, page 1 of 2, requires that RM-G-9 must be available to complete Fuel Shuffles.

The question is at the Memory cognitive level because the operator must recall bits of information to answer the question correctly.

What MUST be known:
1. What automatic actions are associated with the failure of RM-G-9? 2. What automatic actions are associated with the failure of RM-A-4? 3. What automatic actions are associated with the failure of RM-A-8? 4. What automatic actions are associated with the failure of RM-A-14?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	086	K6.04	
	Importance Rating	2.6		

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: RO Question # 36

Plant conditions:

- FS-V-796, FHB Chiller Room is closed for maintenance.
- PLF-4-5, 322' CB Fire PNL Fire is in alarm.
- PLF-4-6, 322' CB Fire PNL Trouble is in alarm.

Subsequently, FS-PNL-717, SBO IONIZATION DETECTOR PANEL, develops a trouble alarm.

Which ONE (1) of the following identifies the Control Room PLF Alarm response to SBO Fire panel trouble?

A. Neither PLF-4-5 NOR PLF-4-6 re-flash.

B. ONLY PLF-4-5 re-flashes.

C. ONLY PLF-4-6 re-flashes.

D. Both PLF-4-5 AND PLF-4-6 re-flash.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Alarm response PLF rev 15 for PLF-4-5 indicates that Fire panel 717 trouble is an input to CB Fire Pnl Fire, PLF-4-6 CB Fire Pnl Trouble indicates that panel 717 fire does not send a signal to PLF-4-6, both panels indicate that with any alarm in the panel will not reflash and that compensatory fire watch is required. Both alarms indicate that fire valve FS-V-796 being closed will bring the alarm in.

B. **Incorrect.** This is plausible because the fire alarm would alarm on Fire panel 717 trouble, if it were not already locked in. Incorrect because the alarm is already in and

will not re-flash for any condition.

C. **Incorrect** This is plausible because the fire alarm would be a logical outcome for a trouble alarm to bring in a trouble alarm on the remote panel. Incorrect because the alarm is not an input and the alarm will not re-flash for any new condition.

D. **Incorrect.** This is plausible the valve closed (a trouble condition) brings in both alarms. However the SBO trouble only goes to the remote panel fire alarm, and neither will re-flash if already in.

Technical Reference(s): Panel Left Front Alarm response procedure rev 15 pgs PLF-4-5 pg 1 of 3, and PLF-4-6 pg 1 of 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 680-GLO-10 and 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

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Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the effect that a loss of a portion of the Fire Protection System will have on the Fire Detection system.

The question is at the Memory cognitive level because the operator must recall the impact of an alarm already in effect has on the detection system.

What MUST be known:
1. The re-flash capability of the Control Building Fire detection system when already

in alarm.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	015	A3.04	
	Importance Rating	3.3		

Ability to monitor automatic operation of the NIS, including: Maximum disagreement allowed between channels

Proposed Question: RO Question # 37

Plant conditions:

- A plant startup is in progress.
- Plant power is 25%.

Which ONE (1) of the following identifies the MAXIMUM allowable meter variance between the Power Range Nuclear Instruments (NI-5, 6, 7 and 8)?

A. ±1%

B. ±2%

C. ±2.5%

D. ±4%

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to TQ-TM-104-623-C001 (p32; Rev 3), practice is to recalibrate NIs when the heat balance error exceeds 1% RP; and the operator may incorrectly believe that this is the maximum allowable meter indication deviation on the Power Range excore detectors. Additionally, according to 1302-1.1 (p6; Rev 58), when calibrating Nis, the power level must be maintained within ±1%, and the operator may incorrectly believe that this is the maximum allowable meter indication deviation on the Power Range excore detectors.

B. **Correct.** According to 1301-1 (p4; Rev 166) When the Reactor is in the "Startup Mode", "Critical", "Hot Standby Condition" or "Power Operation Condition" perform the checks required by Data Sheet 1. According to 1301-1 (p10; Rev 166) twice daily the

operator is required to determine whether or not the power range instrument values all agree within 2% (Via computer points A0582, A0583, A0584 and A0585). This is necessary to satisfy Technical Specification 4.1-1.3 and 3.5-1A.2 when reactor power is > 15%.

C. **Incorrect.** This is plausible because according to TQ-TM-104-623-C001 (p32; Rev 3), Technical Specification 4.1.1, Table 4.1-1 requires the Power Range Nuclear Instruments to be checked and adjusted as necessary each shift to ensure that the power indication is within 2% of actual heat balance reactor power and the offset error (related to imbalance) is less than 2.5%; and the operator may incorrectly believe that this is the maximum allowable meter indication deviation on the Power Range excore detectors.

D. **Incorrect.** This is plausible because according to 1101-1 (p9; Rev 72) and TQ-TM-104-623-C001 (p32; Rev 3), during operation between 15-100 percent power, it is desirable to maintain ± 4 percent power imbalance as measured on the ex-core detectors. The operator may incorrectly believe that this is the maximum allowable meter indication deviation on the Power Range excore detectors.

Technical Reference(s): 1301-1 (p4 and 10; Rev 166)
1302-1.1 (p6; Rev 58)
1101-1 (p9; Rev 72)
TQ-TM-104-623-C001 (p32; Rev 3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-6 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the NIS, specifically, the maximum disagreement allowed between channels.

The question is at the memory cognitive level because the operator must recall bits of information to answer the question.

What MUST be known:
1. At the given power level, what is the maximum allowable meter indication deviation on the Power Range excore detectors.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	034	A4.01	
	Importance Rating	3.3		

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

Proposed Question: RO Question # 38

Plant conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.
- RB Purge is in progress.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Refueling cavity.
- MAP C-1-1 RADIATION HI LEVEL, alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.

No action other than the initiation of a Containment evacuation has taken place.

Which ONE (1) of the following identifies the current positions of RB Purge Isolation Valves (AH-V-1A, 1B, 1C and 1D), AND the reason for their position?

A. Open;
The RB Purge Line Isolation High Radiation Interlock is defeated.

B. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9 when it went above the HIGH level.

C. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9 Hi-Hi when it went above the ALERT level.

D. Open;
The RB Purge Line Isolation High Radiation Interlock has not operated because RM-A-9

Hi-Hi has not exceeded the HIGH level.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to TQ-TM-104-661 (p30-31; Rev 4), RM-A-9 measures the radioactivity of particulates, gas and radioiodine in the RB Purge Exhaust. The gas channel is equipped with a second detector RM-A-9 High-High (RM-G-24) for extended range. According to TQ-TM-104-661 (p53-54; Rev 4), when the RM-A-9 Gas Channel "HIGH ALARM" occurs, it performs three automatic functions: (1) shuts WDL-V-534 and WDL-V-535 (RB sump drain isolation), (2) initiates Automatic Sampling System (MAP 5 Iodine Sampler), and (3) shuts RB purge valves AH-V-1A/1B/1C/1D. This is known as the RB Purge Line Isolation High Radiation Interlock. According to TQ-TM-104-661 (p53-54; Rev 4), interlock signals are provided to minimize plant release or personnel exposure, and they can be defeated in the Control Room. According to 1505-1 (p5; Rev 53) procedures will be in place to require operation of RB Purge Exhaust System and bypass (i.e. defeat) of RB Purge High Radiation Isolation signal whenever irradiated fuel movement is in progress and containment integrity is not maintained. According to 1101.3, Enclosure 3, (p35; Rev 89), this is done because the purge valves must stay open with an RB purge on to prevent RB pressurization. This ensures BWST gravity draining can be accomplished. Consequently, under the stated conditions, the RB Purge Line Isolation High Radiation Interlock is defeated.

B. **Incorrect.** This is plausible because the operator may not be aware of the requirement to defeat the RB Purge High Radiation Isolation signal under the present plant conditions.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the interlock is not associated with RM-A-9, but rather RM-A-9 Hi-Hi; and that the interlock functions when this instrument exceeds the ALERT level. In the plant conditions, the reading of RM-A-9 Hi-Hi has exceeded the ALERT Level, but not the HIGH Level. Additionally, according to TQ-TM-104-661 (p115; Rev 4), revision of the Operator Lesson Plan was made to clarify "confusing Information" related to these two PRM instruments. Apparently, at one time the Lesson Plan was unclear on whether or not RM-A-9 Hi-Hi caused the signal.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the interlock is not associated with RM-A-9, but rather RM-A-9 Hi-Hi; and that the interlock functions when this instrument exceeds the HIGH level. In the plant conditions, the reading of RM-A-9 Hi-Hi has exceeded the ALERT Level, but not the HIGH Level. Additionally, according to TQ-TM-104-661 (p115; Rev 4), revision of the Operator Lesson Plan was made to clarify "confusing Information" related to these two PRM instruments. Apparently, at one time the Lesson Plan was unclear on whether or not RM-A-9 Hi-Hi caused the signal.

Technical Reference(s): TQ-TM-104-661 (p30-31, 53-54, and 70; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor the PRM instrumentation in the control room under a condition when higher than expected radiation levels exist. This is accomplished by presenting the operator with a set of plant conditions in which the expected switch configuration of the identified PRMs must be known, and the implications of these positions understood, to correctly answer the question.

The question is at the Comprehensive/Analysis cognitive level because the operator must recall bits of information, and then evaluate this information against a specific set of plant conditions to correctly answer the question.

What MUST be known:
1. RM-A-9 will automatically close the RB Purge Isolation Valves on the detection of high radiation. 2. A high radiation condition on RM-A-9 exists under the present plant conditions. 3. This interlock has the capability of being defeated, and under certain plant conditions the interlock is defeated. 4. Under the present plant conditions, the interlock is defeated.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E05	EK1.3	
	Importance Rating	3.8		

Knowledge of the operational implications of the following concepts as they apply to the (Excessive Heat Transfer) Annunciators and conditions indicating signals, and remedial actions associated with the (Excessive Heat Transfer).

Proposed Question: RO Question # 39

The Plant was operating at 100% power with ICS in full automatic when the following occurred:

- The Reactor tripped on a low RCS pressure signal.
- OP-TM-EOP-001 IMA's and initial symptom check has been completed.

The following indications are noted:

- RCS pressure is 1800 psig and lowering.
- RCS T_{cold} is 532°F and lowering.
- OTSG 1A pressure is 660 psig and lowering.
- OTSG 1B pressure is 970 psig and steady.
- Pressurizer level is 90 inches and lowering.
- Makeup Tank level is 65 inches and relatively steady.
- Makeup Pump, MU-P-1A is running.
- Reactor Building is 6 psig and rising.

Which ONE (1) of the following identifies the action that is required to be taken <u>FIRST</u> IAW EOP's?

A. Throttle HPI flow IAW Rule 2 per OP-TM-EOP-010, Guide 12.

B. Close Letdown Isolation Valve MU-V-3 per OP-TM-EOP-010, Guide 9.

C. Isolate Emergency Feedwater to OTSG 1A per OP-TM-EOP-010, Rule 3.

D. Isolate Main Steam and Main Feedwater to OTSG 1A per OP-TM-EOP-010, Rule 3.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.**
This is plausible because GUIDE 12 can be directed by either OP-TM-EOP-001 or RULE 3.

However:
- OP-TM-EOP-001 is exited completely prior to initiating Guide 9, and
- RULE 3 has Guide 12 to be initiated AFTER Phase 1 isolation occurs given the conditions provided in the stem.

B. **Incorrect.**
This is plausible because GUIDE 9 can be directed by either OP-TM-EOP-001, OP-TM-EOP-003, or OP-TM-MAP-G0205, PZR LVL HI/LO.

However:
- OP-TM-EOP-001 is exited completely prior to initiating Guide 9,
- OP-TM-EOP-003 has initiation after the step to PERFORM RULE 3, so it is not the first action to take, and
- OP-TM-MAP-G0205 is not the correct path since the question asks "IAW EOPs"

Additionally, MU-V-3 has already closed on the 4# ES actuation.

C. **Incorrect.**
This plausible because according to OP-TM-EOP-010 (p6; Rev 11) Rule 3, Step 4, based on the stated conditions the operator will be directed to PERFORM Phase 2 Isolation of the affected OTSG(s).

IAW OP-TM-EOP-0101 (p19, Rev 6): Phase 2 isolation interrupts Emergency Feedwater supplies to the respective OTSGs and isolates turbine bypass valves (TBV), atmospheric dump valves (ADV), and EF-P-1 steam supply.

However, this will occur **after** the performance of Phase 1 isolation. The operator may incorrectly believe that since there is steam in the RB, this action is given priority.

D. **Correct.**
An analysis of the given conditions will indicate XHT conditions exist and EOP-010, Rule 3 will be entered.

According to OS-24 (p3; Rev 18) Step 3.5, XHT is undesired heat removal by one or both OTSGs.
XHT can be confirmed if ALL of the following conditions exist:
(1) RCS average temperature below 540°F,
(2) Uncontrolled lowering of RCS temperature, and
(3) Tsat for OTSG pressure is less than Tcold on affected OTSG(s).

According to OS-24 (p7; Rev 18) Step 4.1.3.A, prioritization of EOP actions is first based on "symptoms of core cooling upset" (Attachment D). Any time a higher priority symptom is identified, the applicable Rule is performed and that EOP is entered.

The symptom priorities are
(1) Loss of Subcooling Margin,
(2) Excessive Primary-to-Secondary Heat Transfer,
(3) Lack of Primary-to-Secondary Heat Transfer,
(4) and Steam Generator Tube Leak.

Since XHT exists, according to Attachment D (p33; Rev 18), Rule 3, Excessive Heat Transfer (XHT) is applicable.

According to OP-TM-EOP-010 (p6; Rev 11) Rule 3, Step 3, the operator will be directed to PERFORM Phase 1 Isolation of the affected OTSG(s).

IAW OP-TM-EOP-0101 (p18, Rev 6): Phase 1 isolates normal feedwater and steam flow paths to the condenser and atmosphere, but maintains Emergency Feedwater capability and steam alignment to the turbine driven Emergency Feedwater pump.

Technical Reference(s): OS-24 (p3, 7, 16, 33; Rev 19)
OP-TM-EOP-010 (p6; Rev 12) (Attach if not previously provided)

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

Learning Objective: XHT-PCO-1 (As available)

Question Source: Bank # QR-XHT-PCO-1-Q01
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of Excessive Heat Transfer as it applies to the Annunciators and conditions indicating signals, and associated remedial actions. This is accomplished by creating the conditions in which XHT exists, and then requiring the operator to identify the priority of procedural actions.

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate a number of plant conditions and determine its state, and then apply rules of procedure usage to answer the question correctly.

What MUST be known:
1. The conditions for XHT. 2. Whether or not XHT exists in the given conditions. 3. Based on the given conditions, what procedure should be entered.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	007	EK1.02	
	Importance Rating	3.4		

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Shutdown margin

Proposed Question: RO Question # 40

Following a reactor trip:

- All Groups 1 through 7 control rods are verified fully inserted except Group 6 Rod 6.
- Reactor is verified shutdown and neutron flux is lowering as expected.

Which ONE (1) of the following correctly completes the statement below?

Two minutes after the trip adequate Shutdown Margin ___________

A. does exist, and no boration is required.

B. does NOT exist, and no boration is required.

C. does exist, and boration from the BWST must be initiated until Shutdown Margin is verified.

D. does NOT exist, and boration from the BWST must be initiated until Shutdown Margin is verified.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is incorrect because Rule 5 requires an Emergency Boration for any rod not on the bottom in a post-trip situation. According to OP-TM-EOP-001 (p5; Rev 10) Step 3.3, the operator will be required to verify all Rods in Groups 1-7 are fully inserted. if not, the operator is directed to Rule 5. According to OP-TM-EOP-010 (p10; Rev 11), if the Reactor is shutdown and all control rods are not fully inserted, which is the case in the stated conditions, then Emergency Borate until 1% dk/k SHUTDOWN has been achieved for the expected plant condition IAW Figure 1 of 1103-4, "Soluble

Poison Concentration Control", or 1103-15A, "SDM and Reactivity Balance." This is plausible because the operator may correctly believe that the required SDM is met, and because of that Rule 5 is NOT required to be initiated.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that with one Control Rod fully withdrawn the required SDM does NOT exist, but that Emergency Boration is not required due to the Reactor being shutdown and neutron flux lowering as expected.

Rule 5 entry criteria:
IAAT any of the following conditions exist:

- Emergency boration is directed by procedure,
- Reactor is shutdown and all control rods are not fully inserted,
- Reactor is shutdown and Neutron flux is not lowering as expected,

The candidate may believe that the entry criteria is an ALL of the following conditions.

C. **Correct.** In the event a control rod fails to insert for a reactor trip, there is sufficient negative reactivity for tripping the reactor and maintaining a hot shutdown condition. This requirement is met as a criteria of the core design. All B&W reactor cores are required to be capable of maintaining 1 percent delta-k/k shutdown margin at hot shutdown conditions with the maximum worth control rod withdrawn from the core. According to OP-TM-300-205, Attachment 7.2, (p8; Rev 2), the Shutdown Margin at Hot Shutdown - Qualitative Assessment Instruction Sheet, all inoperable rods must be fully inserted, except that one inoperable rod may be fully withdrawn. In other words, there is no penalty against SDM if one Control Rod is fully withdrawn. However, according to OP-TM-EOP-010 (p5; Rev 10) Step 3.3, the operator in a post-trip situation will be directed to VERIFY control rod groups 1through 7 are fully inserted; and if not, INITIATE Rule 5, "Emergency Boration. According to OP-TM-EOP-010, Rule 5, (p10; Rev 11), the operator will be directed to initiate Emergency Boration until such time as a SDM of 1% dk/k SHUTDOWN can be verified by plant procedures.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that with one Control Rod fully withdrawn the required SDM does NOT exist, and that boration is required via the BWST.

Technical Reference(s):	OP-TM-300-205, Attachment 7.2, (p8; Rev 2) OP-TM-EOP-010 (p5; Rev 10) OP-TM-EOP-010, Rule 5, (p10; Rev 11) OP-TM-EOP-001 (p5; Rev 10)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:		EOP-001-PCO-4	(As available)
Question Source:	Bank #	IR-EOP-001-PCO-4-Q03	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	None

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of shutdown margin. This is accomplished by requiring that the operator know that in a post-trip situation, the core is designed to maintain SDM, even with one control rod fully withdrawn.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. Is SDM maintained in a post-trip situation, even with one Control Rod fully withdrawn? 2. If in a post-trip situation, and only one control rod is fully withdrawn, does the operator need to initiate Rule 5?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008 AK1.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure

Proposed Question: RO Question # 41

Plant conditions:

- The crew has entered the EOP for a Pressurizer Steam Space break.
- RCS pressure stabilizes at 1800 psig.
- The leak estimate is approximately 600 gpm.

Subsequently:

- The crew initiates a LOCA Cooldown in accordance with Guide 11, Cooldown Rate Limits.
- 30 minutes after the cooldown is initiated the RCS pressure is 1200 psig.

Which ONE (1) of the following approximates the current break flow, AND the Pressurizer level?

A. Approximately 390-410 gpm ;
Pressurizer level will be off-scale high.

B. Approximately 390-410 gpm ;
Pressurizer level will be lowering.

C. Approximately 480-500 gpm ;
Pressurizer level will be lowering.

D. Approximately 480-500 gpm ;
Pressurizer level will be off-scale high.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may

mis-understand the concept of break flow being proportional to the ΔP. If so, the operator may conclude that since the present ΔP is about 2/3 of the original ΔP, then the break flow should be about 2/3 of the original break flow. If so, the operator will conclude that break flow is approximately 390-410 gpm.

B. **Incorrect.** 1st part wrong, 2nd part wrong. See A and C.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the cooldown will cause the Pressurizer level to lower.

D. **Correct.** 1st part correct, 2nd part correct. Leak rate is proportional to the square root of the ΔP. As pressure lowers, so will leak rate at a decreasing rate. The current break flow may be determined as follows: It is known that there is 600 gpm with a ΔP of ≈1800 psig. The square root of 1800 = 42.43, which is proportional to 600 gpm. The current ΔP is ≈1200 psig, and the square root of 1200 is 34.64. With these numbers the operator can solve for the current break flow by multiplying 600 gpm x 34.64/42.43 ≈ 489 gpm; or approximately 480-500 gpm. According to TM1 UFSAR (p14.2-36; Rev 20) the system pressure transient for a small break in the pressurizer will behave like other SBLOCAs. The initial depressurization, however, will be more rapid as the initial inventory loss is entirely in the form of steam. The pressurizer level response for these accidents will initially behave like a very small break without auxiliary feedwater. The initial rise in pressurizer level shown on Figure 14.2-51 will occur due to the pressure reduction in the pressurizer and an insurge of coolant into the pressurizer from the RCS. Once the reactor trips, system contraction causes a decreasing level in the pressurizer. Flashing will ultimately occur in the hot leg piping and cause an insurge into the pressurizer. This ultimately fills the pressurizer. For the remainder of the transient, the pressurizer will remain full.

Technical Reference(s): TM1 UFSAR (p14.2-36; Rev 20) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP006-PCO-5/EOP002-PCO-5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of the Pressurizer Vapor Space Accident, specifically that a change in leak rate will occur with a change in pressure

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate an understanding of fluid flow concepts to correctly answer the question.

What MUST be known:
1. How will break flow be affected by a pressure drop to 2/3 of the original pressure of the leaking system? 2. How does Pzr level respond to a Vapor Space Leak.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 EK2.06	
	Importance Rating	2.9	

Knowledge of the interrelations between the following and ATWS: Breakers, relays, and disconnects

Proposed Question: RO Question # 42

Which ONE (1) of the following describes the minimum input for Diverse Scram System actuation which will generate a reactor trip signal AND its response following this actuation?

A. One Safety Grade RCS Pressure Wide Range detector is > 2500 psig; AND
The shunt trips for 1G and 1L breakers are energized from 120 VAC tripping the reactor.

B. Both Safety Grade RCS Pressure Wide Range detectors are > 2500 psig; AND The shunt trips for 1G and 1L breakers are energized from 120 VAC tripping the reactor.

C. One Safety Grade RCS Pressure Wide Range detector is > 2500 psig; AND
The Pulse Generators are disabled de-energizing the Single Rod Power Supplies (SRPS) modules tripping the reactor.

D. Both Safety Grade RCS Pressure Wide Range detectors are > 2500 psig; AND
The Pulse Generators are disabled de-energizing the Single Rod Power Supplies (SRPS) modules tripping the reactor.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. See B and C.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because up until 2011 this is how the DSS operated. A modification was made to the Control Rod Drive System that impacted this function. The operator may be unaware of this change, or have incorrect knowledge of its impact.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because one Safety Grade RCS Pressure Wide Range detector > 2500 psig will cause a single train trip,

and the operator may incorrectly believe that it causes a dual train trip.

D. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-MAP-G0207 (p1; Rev 1), if RCS pressure is > 2500 psig (one or both trains, from RC-PT-949 and/or RC-PT-963) a DSS actuation of some sort will occur. One RCS Wide Range pressure instrument in excess of 2500 psig will result in a Single Train Trip, while both RCS Wide Range pressure instruments in excess of 2500 psig will result in a Dual Train Trip. Also according to OP-TM-MAP-G0207 (p1; Rev 1), a Single Train Trip will result in the pulse generators of one train being disabled which de-energizes one train of SRPS modules (and reduces the Control Rod Drive system to single train operation). Also this action trips the RTBs on that train. Likewise, if a Dual Train Trip occurs, all Pulse Generators are disabled which de-energizes all SRPS modules and trips the reactor. This also trips all four RTBs. Consequently, the minimum input for Diverse Scram System actuation which will generate a reactor trip signal is Both Safety Grade RCS Pressure Wide Range detectors are > 2500 psig AND its response following this actuation is to disable the Pulse Generators which will de-energize the SRPS modules which will trip the reactor.

Technical Reference(s): OP-TM-MAP-G0207 (p1; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-2 (As available)

Question Source: Bank #

Modified Bank # WTSI 62758 (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between the Reactor Trip Breakers, RPS Relays and CRDM disconnects with respect to ATWS protection.

The question is at the Memory cognitive level because the operator must recall specific bits of information to correctly answer the question.

What MUST be known:
1. A modification has occurred to the CRDM System which has changed the means by which the DSS performs its function. 2. A means of the DSS performing its function is no longer tripping of 1G and 1L breakers, but rather disabling all Pulse Generators which de-energizes the SRPS modules and trips the reactor. 3. The DSS will generate a reactor trip signal on a dual train trip when signals from two Wide Range RCS pressure instruments are > 2500 psig. 4. The DSS will NOT generate a reactor trip signal on a single train trip when a signal from one Wide Range RCS pressure instrument is > 2500 psig.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015 AK2.10	
	Importance Rating	2.8	

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls

Proposed Question: RO Question # 43

Initial plant conditions:

- 75% power
- ICS in Automatic

Subsequently:

- RC-P-1B motor amps lowers to 380 amps.
- MAP F-3-1, RC LOOP A FLOW LO is LIT.
- MAP F-1-1, RCP MOTOR TRIP is NOT lit.

Which ONE (1) of the following identifies the automatic action(s), if any, that have occurred?

A. No automatic actions.

B. A 50%/minute runback occurs, ONLY.

C. A feedwater flow re-ratio occurs, ONLY.

D. A 50%/minute runback and feedwater flow re-ratio occurs.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that both the runback and the re-ratio signals are tieed to the RCP breaker position which has not changed.

B. **Incorrect.** This is plausible because according to OP-TM-MAP-H0101 (p1; Rev 1) there are two 50%/minute runbacks associated with the loss of one or more RCPs, one

to 50% load and a second to 75% load. However, according to TQ-TM-104-621-C001 (p28; Rev 2), the input for these runbacks comes from the RCP breaker position (52 contacts), and the breakers have NOT opened. The operator may incorrectly believe that the runback is occurring, while the reratioing requires the RCP Breaker to be OPEN. Additionally, according to TQ-TM-104-621-C001 (p30; Rev 2), there is a variable low RC flow runback that may occur under the stated conditions (i.e. NOT from RCP Breaker position), however, this occurs at 20%/minute.

C. **Correct.** RC-P-1B has had a reduction in flow as indicated by the reduction in pumps amps. Because the flow reduction was in only one RC loop, the loop Tc will change. This will cause the ΔTc circuit to re-ratio feedwater to return the loop Tc to the same value. According to TQ-TM-104-621-C001 (p40-41, 145-146; Rev 2), The ΔTc control to the ratio circuit is in two parts: (1) through a selector station, which gives the operator the means of establishing a manual ratio, ΔTc is applied by changing the gain of the demand signal to one steam generator and subtracting it from the total demand. This resulting difference represents the demand to the second steam generator. (2) RC Flow Error, the second part of the ΔTc control is a feed forward circuit, which anticipates a ΔTc error upon the loss of a reactor coolant pump. If one of the reactor coolant pumps were to fail, the reactor coolant flow in that loop would drop. With a greater primary flow in one loop than the other and the same feedwater flow in both loops, the Tc of the loop with the low primary flow would be much less than that of the other loop. The only way to maintain ΔTc equal to setpoint is to re-ratio the feedwater flow in each loop to correspond to the new primary flow ratio. This action is accomplished through the use of the reactor coolant flow error. The difference between the reactor coolant flow in the two loops is applied to the ratio control circuit. By applying this action to the Feedwater ratio control, the sum of the loop feedwater flows will be maintained equal to the unit demand. According to OP-TM-MAP-F0301 (p1; Rev 0), the operator is directed to ensure that the feedwater is re-ratioing.

D. **Incorrect.** This is plausible because the 2nd part is correct. However, the 1st part is wrong (See B).

Technical Reference(s): TQ-TM-104-621-C001 (p40-41, 145-146; Rev 2)
OP-TM-MAP-F0301 (p1; Rev 0)
OP-TM-MAP-H0101 (p1; Rev 1)
OP-TM-MAP-F0101 (p1; Rev 1)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-5 (As available)

Question Source: Bank #

Modified Bank #		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between the Reactor Coolant Pump malfunctions (Loss of RC Flow) and the RCP indicators and controls. This is accomplished by presenting the operator with the status of two MAPs and the RCP motor amp indication, and requiring the operator to identify the automatic actions that should be occurring at the time.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information and apply this information to a given set of plant conditions, to predict plant response, in order to correctly answer the question.

What MUST be known:
1. What are the normal running amps for the RCP at 100% power? 2. What does it mean when one RCP has substantially lowered amps? 3. How does the ICS respond to the trip of one RCP at 75% power? 4. What causes the ICS to re-ratio feedwater flow?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011 EK2.02	
	Importance Rating	2.6	

Knowledge of the interrelations between the following Large Break LOCA: Pumps

Proposed Question: RO Question # 44

Initial conditions:

- Time = 0100
- Reactor tripped from 100% power
- RCS pressure = 45 psig
- Core Subcooled Margin Meters indicate 0°F

Current conditions:

- Time = 0102
- Core Subcooled Margin Meters indicate 0°F
- ALL Reactor Coolant Pumps are operating
- Reactor Coolant Pump amps are fluctuating
- LPI flow in both trains is 1200 gpm.
- An operator is beginning to perform Rule 1, Loss of Subcooling Margin

Which ONE (1) of the following identifies the action that the operator will take with the Reactor Coolant Pumps, and the reason for this action?

A. Stop the RCPs because they will be severely damaged if left running.

B. Stop the RCPs to prevent subsequent core damage that would result if the pumps were stopped at an inopportune time.

C. Leave RCPs operating because there is insufficient ECCS flow being delivered to the core, and the RCPs may be able to supplement fluid flow into the core.

D. Leave RCPs operating because this accident will result in the implementation of a Severe Accident Management Guideline, which assumes that the RCPs are operating.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator will diagnose that the pump amps fluctuating the RCPs will be damaged. However, the operating directed to leave the RCPs running if not turned off within one minute of the LSCM, even if the pumps are damaged.

B. **Incorrect.** This is plausible because this is the basis for stopping them within one minute of the LSCM in the SBLOCA event. If the RCPs were stopped when the RCS void fraction has progressed to ≥70%, the resulting loss of inventory could cause the core to substantially uncover and be damaged.

C. **Correct.** According to OP-TM-EOP-0101 (p6; Rev 6), the intent of Step 2 of Rule 1 is to stop forced flow. However, there are some considerations. RCPs are tripped immediately to prevent possible core damage during specific size small break loss of coolant accidents (LOCA). Core damage could occur if the RCPs trip later in the event. Because RCP trip cannot be eliminated as a possibility, RCP trip is performed as soon as the loss of subcooling margin is recognized. This precludes the possibility of RCP trip when the RCS void fraction has progressed to the point (approx. 70% void fraction) where RCP trip would result in a collapsed liquid level below the top of the core. In addition, once RCPs are tripped, the rate of inventory loss is reduced to the point where HPI (with primary-to-secondary heat transfer for smaller breaks) can remove decay heat and maintain the core covered. Operationally, RC-P-1A/B/C/D should be immediately secured from the console. If any of the pumps are unable to be secured from the console, the 1A/1B 6900v busses should be immediately de-energized. If the operator is unable to secure all 4 RCPs within a minute of the LSCM, all running RC pumps must remain running, even though RCP damage is possible, until specific shutdown criteria is satisfied. In the stated conditions the LSCM has existed for 2 minutes, and the RCPs are all running. Since the operator was unable to secure all 4 RCPs within a minute of the LSCM, all running RC pumps must remain running, even though RCP damage is possible, until specific shutdown criteria is satisfied. The specific shutdown criteria are (1) > 25°F of subcooling, (2) LPI flow > 1250 gpm or (3) if Tclad is > 1800°F. The fact that Subcooling and total LPI flow are low indicate that there is insufficient ECCS flow into the core to keep it covered, and the RCPs should remain running, even though they may be damaged, to ensure that whatever liquid is available in the RCS can be pumped into the core area.

D. **Incorrect.** This is plausible because according to OP-TM-EOP-0101 (p6; Rev 6), one of the three criteria for turning the RCPs off after Step 2 is if Tclad is > 1800°F. The basis for this is that a Tclad of > 1800 °F signifies a "Severe Accident" and requires transfer to the Technical Support Center (TSC) Severe Accident Management Guidelines (SAMG). The guidelines assume the reactor coolant pumps are not operating upon entry. The operator may confuse the concepts.

Technical Reference(s): OP-TM-EOP-0101 (p6; Rev 6) (Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	EOP010-PCO-4		(As available)
Question Source:	Bank #	WTSI 60230	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	2005
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the operator must demonstrate Knowledge of the interrelations between the Large Break LOCA and the Reactor Coolant Pumps. This is accomplished by presenting the operator with a set of conditions that would be typical of a Large Break LOCA, and then requiring that they identify what action is required regarding the RCPs, and the basis for the action.

The question is at the Comprehension/Analysis cognitive level because the operator must the operator must demonstrate understanding by indicating why an action is taken to correctly answer the question; additionally, the question is at the Comprehension/Analysis cognitive level because the operator must evaluate the plant conditions over time, and apply the rules for RCP pump trip to correctly answer the question.

What MUST be known:
1. How does Rule 1 address the RCPs in the first minutes after the loss of SCM occurs? 2. Why is rule 1 written the way that it is (i.e. what is the basis for the ARCP trip rules? 3. Under the stated plant conditions what must be done with the RCPs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 EK3.01	
	Importance Rating	2.7	

Knowledge of the reasons for the following responses as the apply to the Station Blackout: Length of time for which battery capacity is designed

Proposed Question: RO Question # 45

IAW TMI Unit 1 Updated Final Safety Analysis Report (UFSAR), which ONE of the following identifies:
(1) The length of time that the Station Batteries are designed for a Station Blackout Event, and
(2) A criterion upon which the time capacity in part (1) for the Station Batteries is based?

A. (1) 2 Hours continuously
(2) To provide emergency lighting in vital areas to support performance of time critical tasks.

B. (1) 2 Hours continuously
(2) To support remote control of emergency feedwater flow to the OTSGs during plant cooldown.

C. (1) 4 Hours continuously
(2) To provide emergency lighting in vital areas to support performance of time critical tasks.

D. (1) 4 Hours continuously
(2) To support remote control of emergency feedwater flow to the OTSGs during plant cooldown.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**
Part 1 is correct.

Part 2 is incorrect but plausible because there are emergency DC lights in vital areas to support accomplishment of time critical actions. However, this is not a basis for battery time capacity. The operator may incorrectly believe that lighting to perform time critical

actions is essential and that this criterion forms the basis for the time capacity of the Station batteries.

B. **Correct.**
Part 2 is correct: According to TQ-TM-104-734-C001 (p27-28; Rev 6), OP-TM-AOP-0231 (p1; Rev 1) and OP-TM-AOP-0241 (p1; Rev 1), the capacity of each of the two redundant batteries is sufficient to feed its connected essential load for 2 hours continuously and perform three complete cycles of safeguard breaker closures and subsequent tripping. The 2 hour rating is based on the time required to ensure that all nuclear and BOP emergency equipment can perform its intended function and on the criteria contained in the Institute of Electrical and Electronics Engineers (IEEE) standards for Class 1E electrical systems.

Part 2 is correct: According to TQ-TM-104-424-C001 (p28-29; Rev 8), the controls and auxiliary systems for the emergency feed pump operate on DC power from the battery backed DC bus for a minimum of two hours.

C. **Incorrect.**
Part 1 is incorrect. 2 Hours is the rating. Four hours is the duration of the FSAR Station Blackout Evaluation. Plausible if the examinee believes the Station battery will be available for the duration of the evaluation.

Part 2 is incorrect but plausible because there are emergency DC lights in vital areas to support accomplishment of time critical actions. However, this is not a basis for battery time capacity. The operator may incorrectly believe that lighting to perform time critical actions is essential and that this criterion forms the basis for the time capacity of the Station batteries.

D. **Incorrect.**
Part 1 is incorrect. 2 Hours is the rating. Four hours is the duration of the FSAR Station Blackout Evaluation. Plausible if the examinee believes the Station battery will be available for the duration of the evaluation.

Part 2 is correct: According to TQ-TM-104-424-C001 (p28-29; Rev 8), the controls and auxiliary systems for the emergency feed pump operate on DC power from the battery backed DC bus for a minimum of two hours.

Technical Reference(s):	TQ-TM-104-734-C001 (p27-28; Rev 6) TQ-TM-104-424-C001 (p28-29; Rev 8) OP-TM-AOP-0231 (p1; Rev 1) OP-TM-AOP-0241 (p1; Rev 1) TMI-1 UFSAR (p8.2-6, Rev 20)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	734-GLO-7		(As available)
Question Source:	Bank #		
	Modified Bank #	WTSI 44106/IR-734-GLO-7-Q02	(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	2003
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	8	
	55.43		

Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the operator must demonstrate knowledge of the reasons for the length of time for which battery capacity is designed as it applies to the Station Blackout.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. The capacity of each of the two redundant batteries is sufficient to feed its connected essential load for 2 hours. 2. The controls and auxiliary systems for the emergency feed pump are essential for this two hour period following an SBO.

Original Question:

According to the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR), which ONE (1) of the following identifies a criterion upon which the time capacity for the Station Batteries is based, for the station blackout event?
A.Maintain RCP lift oil pressure during pump coastdown.
B.Provide for turbine-generator bearing oil flow during shaft coastdown.
C.Provide emergency lighting in vital areas to support performance of time critical tasks.
D.Support remote control of emergency feedwater flow to the OTSGs during plant cooldown.

Proposed Answer:D

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E04	EK3.4	
	Importance Rating	3.5		

Knowledge of the reasons for the following responses as they apply to the (Inadequate Heat Transfer) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Proposed Question: RO Question # 46

Plant conditions:

- LOHT has occurred.
- RCS temperature = 585°F and rising slowly.
- Core SCM = 31°F and slowly lowering.
- The crew is implementing OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer, Attachment 1, OTSG Feed Using a Condensate Booster Pump.
- 1A OTSG level is 8" Startup Range and stable.
- 1B OTSG level is 7" Startup Range and stable.

Which ONE (1) of the following identifies the action required, AND the reason for this action?

A. Reduce and maintain OTSG pressure at 500-600 psig;
This allows continuous Condensate Booster Pump flow and ensures Tube to Shell Delta Temperature will be limited.

B. Reduce and maintain OTSG pressure at 500-600 psig;
This ensures no cavitation damage will occur to the Main Feedwater Nozzles, and ensures Tube to Shell Delta Temperature limits will not be exceeded.

C. Reduce and maintain OTSG pressure at 650-750 psig;
This ensures no cavitation damage will occur to the Main Feedwater Nozzles, and ensures Tube to Shell Delta Temperature limits will not be exceeded.

D. Reduce and maintain OTSG pressure at 650-750 psig;
This allows continuous Condensate Booster Pump flow and ensures Tube to Shell Delta Temperature will be limited.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-004, Attachment 1, (p11; Rev 7), Step 7 the operator is directed to throttle MS-V-3's to maintain 500-600 psig in the OTSGs. According to OP-TM-EOP-0041 (p9; Rev 2) the TBVs/ADVs are utilized to lower OTSG pressure to < 600 psig to allow for continuous CBP flow. Maintaining > 500 psig will limit the TSDT.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to OP-TM-EOP-0041 (p9; Rev 2) maintaining > 500 psig will limit the TSDT, but not necessarily prevent it from exceeding the established limits. This is why the operator is directed to maintain TSDT IAW Guide 14. Additionally, according to OP-TM-EOP-0041 (p9; Rev 2) a minimum MFW flow (0.16Mlb/hr) is desired to ensure the main feedwater nozzles remain full and to prevent cavitation type damage to the Main Feedwater nozzles if they are not full of subcooled fluid. The operator may incorrectly believe that these two statements, found in the basis section of the Attachment, forms the basis for the pressure band.

C. **Incorrect.** 1st part wrong, 2nd part wrong. See B and D.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to OP-TM-EOP-004 (p11; Rev 7) Step 4, the operator is directed to defeat the OTSG Lo-Lo Pressure MFW Isolation when OTSG pressure is < 750 psig. The operator may incorrectly believe that the OTSG pressure band to be maintained is 650-750 psig.

Technical Reference(s): OP-TM-EOP-004, Attachment 1, (p11; Rev 7) OP-TM-EOP-0041 (p9; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate Knowledge of the reasons for the RO functions of feeding the OTSGs with the CBPs, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated as it relates to following responses as they apply to the Inadequate Heat Transfer. Specifically, the operator must identify a pressure band for maintaining OTSG pressures during an LOHT where the CBPs are used as the source of feedwater, and then identify the reasons for the pressure band.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question correctly.

What MUST be known:
1. What is the pressure band for the OTSGs when attempting to fill in an LOHT situation with the CBPs? 2. What is the discharge pressure of the CBPs? 3. Why is this pressure band required?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	EK3.12
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as the apply to the small break LOCA: Letdown isolation

Proposed Question: RO Question # 47

Plant conditions:

- 5% power.

Subsequently:

- The crew has entered OP-TM-AOP-050, Reactor Coolant Leakage, due to a suspected leak in the reactor building.
- The following parameters are observed:

Makeup Tank Level:	50 inches and lowering
Letdown flow:	45 gpm and stable
Letdown Temperature:	125°F and stable
Makeup flow:	225 gpm and stable
Seal Injection Flow:	32 gpm and stable
Pressurizer Level:	95 inches and slowly lowering

The CRS then directs that MU-V-3, Letdown Isolation Valve, be CLOSED.

Which ONE (1) of the following identifies the reason that MU-V-3 was CLOSED?

A. RCS inventory control.

B. The leak was on the Letdown Line.

C. As a backup to the letdown line high temperature interlock.

D. To protect the higher groups of Pressurizer heaters from subsequently overheating.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-AOP-050 (p3; Rev 1), upon entry into the RCS Leakage AOP, the operator will be directed to initiate OP-TM-EOP-010, Guide 9, "RCS Inventory Control". According to OP-TM-EOP-010, Guide 9 (p20-21; Rev 11) the operator will be directed to evaluate MU Tank Level Control, and then Pressurizer Level Control. Since the MU Tank is 50 inches and lowering, this is consistent with an RCS Leak. According to OP-TM-EOP-010 (p20; Rev 11), when the reactor is critical, "desired" pressurizer level is 200 to 240 inches. According to OP-TM-EOP-010 (p21; Rev 11), a series of steps is provided to evaluate the Pressurizer Level control. The operator is first directed to close MU-V-5 so that letdown flow is adjusted to normal flow (i.e. 45 gpm). Next makeup flow is increased in a series of steps that opens further MU-V-17, and if insufficient flow is obtained, its bypass, MU-V-217. A flowrate of 225 gpm is consistent with MU-V-217 having been opened. IF under these conditions, Pressurizer Level is still lowering, the operator is directed to close MU-V-3. According to OP-TM-EOP-0101 (p41; Rev 3), Step C7, after introducing additional makeup with MU-V-217, the effect on pressurizer level is evaluated. The time spent evaluating should be commensurate with the variance of pressurizer level from the desired trend. If pressurizer level is not being restored (i.e. pressurizer level is not rising) after the effects of MU-V-217 have been considered then close MU-V-3 to isolate letdown. Since MU-V-3 has just been closed, the valve was closed for RCS Inventory control.

B. **Incorrect.** This is plausible because according to OP-TM-AOP-050 (p5; Rev 1) Step 3.13.2, if isolation of Letdown is required to stop the leak, then GO TO Section 5.0, "Isolation Of Letdown". The operator may incorrectly believe that the leak is on the Letdown line.

C. **Incorrect.** This is plausible because according to OP-TM-AOP-050 (p3; Rev 1), IAAT letdown temperature > 145 °F, then ENSURE MU-V-2A and MU-V-2B are CLOSED. According to OP-TM-AOP-0501 (p5; Rev 1) the reason for this is to backup to the letdown line high temperature interlock.

D. **Incorrect.** This is plausible because the Pressurizer Heaters will uncover if left unattended. However, according to TQ-TM-104-220 (p8; Rev 5) the heaters start to uncover at 63.5 inches Pressurizer level indication. The operator may incorrectly believe that they will uncover earlier, and/or the operator may incorrectly believe that the main reason is to protect the Pzr Heaters and not keeping the core covered.

Technical Reference(s):	OP-TM-AOP-050 (p3 and 5; Rev 1) OP-TM-AOP-0501 (p5; Rev 1) OP-TM-EOP-010, Guide 9 (p20-21; Rev 11) OP-TM-EOP-0101 (p40; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	AOP5-PCO-1, 5 and 6		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the reasons for letdown isolation during a small break LOCA.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information in AOPs/EOPs and apply these to a given set of plant conditions; and then demonstrate an understanding of why a particular action was taken, to correctly answer the question.

What MUST be known:
1. Normal parameters for Letdown flow, temperature, MU Tank level, MU flow and Seal Injection flow. 2. Know how the above parameters would be expected to change for a RCS leak in the RB. 3. Know the potential causes of isolating letdown during a leak of RCS. 4. Know the means of isolation of Letdown for the various reasons. 5. Given the plant conditions, why was letdown isolated. 6. Know that MU-V-5 is closed during Guide 9 with a low Pzr level to reduce letdown to normal flow, and form the basis of evaluating the Pzr level control system performance?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054	AA1.02
	Importance Rating	4.4	

Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): Manual startup of electric and steam-driven AFW pumps

Proposed Question: RO Question # 48

Plant Conditions:

- The Control Room has been evacuated due to an explosion and subsequent fire.
- Main Feedwater Pumps have been tripped.
- Emergency Feedwater Pumps are not running, and could not be started from the Control Room.

Which ONE (1) of the following correctly completes the statement below?

The Turbine Drive Emergency Feedwater Pump, EF-P-1, may be started ____(1)____, and the Motor Drive Emergency Feedwater Pumps, EF-P-2A/B, may be started ____(2)____.

A. (1) remotely from the RSD panel
(2) remotely from the RSD panel

B. (1) remotely from the RSD panel
(2) remotely from the ES switchgear

C. (1) locally at the pump
(2) remotely from the RSD panel

D. (1) locally at the pump
(2) remotely from the ES switchgear

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-424-C001 (p23; Rev 8) EF-V-30A through D can be operated from the RSD Panels; and the operator may incorrectly believe that EF-P-1 can be operated from there as well.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to TQ-TM-104-424-C001 (p23; Rev 8) EF-V-30A through D can be operated from the RSD Panels; and the operator may incorrectly believe that EF-P-2A/B can be operated from there as well.

D. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-424-C001 (p21; Rev 8) EF-P-1 can be started locally at the pump. According to OP-TM-EOP-020 (p9; Rev 13) Step 3.6 RNO, the Motor Driven EFW Pumps can be started and stopped at the respective 4160 V Bus.

Technical Reference(s):	TQ-TM-104-424-C001 (p21; Rev 8) OP-TM-EOP-020 (p9; Rev 13)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	424-GLO-12	(As available)

Question Source:	Bank #	IR-424-GLO-12-Q01	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate the ability to conduct the manual startup of electric and steam-driven AFW pumps this applies to the Loss of Main Feedwater (MFW).

The question is at the Memory cognitive level because the operator must recall the locations from where the EFW Pumps can be started other than from the control room.

What MUST be known:
1. The alternate location for staring EF-P-1 if the control room is not available. 2. The alternate location for staring EF-P-2A/B if the control room is not available.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056	AA1.04
	Importance Rating	3.2	

Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: Adjustment of speed of ED/G to maintain frequency and voltage levels

Proposed Question: RO Question # 49

Plant conditions:

- The plant has tripped from 100% power due to a Loss of Offsite Power (LOOP).
- The crew has implemented OP-TM-AOP-020, Loss of Station Power.

The following conditions are observed:

- EG-Y-1A Frequency 61.2 Hz
- 1D 4160V Bus Voltage 4120 v
- EG-Y-1B Frequency 59.4 Hz
- 1E 4160V Bus Voltage 4260 v

IAW OP-TM-861-901/902, which ONE (1) of the following identifies the Emergency Diesel Generator(s) that must be adjusted by the operator?

A. Adjust Emergency Diesel Generator EG-Y-1A Voltage Rheostat.

B. Adjust Emergency Diesel Generator EG-Y-1B Voltage Rheostat.

C. Adjust Emergency Diesel Generator EG-Y-1A Speed Governor.

D. Adjust Emergency Diesel Generator EG-Y-1B Speed Governor.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because of the operator may incorrectly believe that the voltage of EG-Y-1A is outside the allowable specification range; and that all other parameters are within the allowable specification range.

B. **Incorrect.** This is plausible because of the operator may incorrectly believe that the voltage of EG-Y-1B is outside the allowable specification range; and that all other parameters are within the allowable specification range.

C. **Correct.** According to OP-TM-AOP-020 (p3; Rev 15), the operator will be directed to initiate both OP-TM-861-901, EG-Y-1A Emergency Operations, and OP-TM-861-902, EG-Y-1B Emergency Operations. According to OP-TM-861-901 (p2; Rev 12) Step 4.2, the operator will address Section 4.3 because EG-Y-1A is loaded onto the 1D 4160V Bus. According to OP-TM-861-901 (p3; Rev 12) Step 4.3.3 and 4, the operator will be directed to adjust the EG-Y-1A governor so that frequency is maintained between 59 and 61 Hz; and adjust voltage to maintain between 4100-4300V.
According to OP-TM-861-902 (p3; Rev 12) Step 4.3.3 and 4, the operator will be directed to adjust the EG-Y-1B governor so that frequency is maintained between 59 and 61 Hz; and adjust voltage to maintain between 4100-4300V. Since frequency is high, EG-Y-1B governor must be adjusted to return frequency to within band parameters.

D. **Incorrect.** This is plausible because of the operator may incorrectly believe that the frequency of EG-Y-1B is outside the allowable specification range; and that all other parameters are within the allowable specification range.

Technical Reference(s): OP-TM-AOP-020 (p3; Rev 15)
OP-TM-861-901 (p2-3; Rev 12)
OP-TM-861-902 (p2-3; Rev 12) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to monitor the parameters necessary to make needed adjustments of EDG speed and voltage as required to maintain needed frequency and bus voltage during a loss of off-site power.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. During a loss of off-site power what frequency range must be maintained on the EDGs? 2. During a loss of off-site power what range of Bus voltage must be maintained on the emergency buses?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AA1.01
	Importance Rating	4.0	

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs

Proposed Question: RO Question # 50

Plant conditions:

- 100% power.
- ICS is in AUTO.
- RCS pressure is 2155 psig and steady.

The following event occurs:

- Pressurizer heater pressure control SETPOINT (RC3-PIC) signal fails to zero (0) psig.

Which ONE (1) of the following identifies the required actions?

A. Place the Diamond Rod Control station in manual and insert control rods.

B. Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1.

C. Verify all pressurizer heaters are de-energized OR manually de-energize heaters.

D. Adjust Heater Banks 1, 2 & 3 using Pressurizer Pressure Bailey Control Station in hand.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because if the analysis of the transient were to mistakenly lead to the determination that ALL the requirements for entry into OP-TM-AOP-070 were met, this would be the correct action. According to OP-TM-AOP-070 (p1; Rev 3) this procedure is entered if the reactor is not shutdown, RCS pressure is not being controlled, and Tavg is rising with RCS pressure > 2205 psig, or Tavg is lowering with

RCS pressure < 2105 psig. If the operator incorrectly entered this procedure, the required action is to ensure diamond station in MAN and INSERT control rods as necessary to reduce power below Reactor power limit and for gross balance with total FW flow.

B. **Incorrect.** This is plausible because if the analysis of the failure causes a belief the PZR pressure will lower uncontrollably and the Reactor trip setpoint will be reached rapidly, the operator may take this action to prevent a reactor trip. However, the proper analysis will show that this failure has only caused the de-activation of three heater banks and the transient is very controllable.

C. **Incorrect.** This is plausible if the analysis of the failure causes a belief the PZR pressure will rise uncontrollably.

D. **Correct.** According to TQ-TM-104-220-C001 (p8-9; Rev 5), there are five banks of heaters containing 13 groups of Pressurizer Heaters. Banks 1 through three, which control groups 1 through 6, are controlled by RC3-PIC. The other two banks, which control groups 7-13 are controlled by Bistables. If RC3-PIC control SETPOINT signal fails to zero (0) psig, the automatically controlled heaters will de-energize because RCS pressure is greater than setpoint. According to OS-24 (p13; Rev 18) Step 4.2.A, Licensed operators may take action without procedural guidance, and without taking a variance the action is taken to directly compensate for the failure of an automatic system. Therefore, the operator will without direction place the Pzr Controller in HAND per OS-24. Once the operator is controlling Pressurize pressure manually, OP-TM-220-503 applies. According to OP-TM-220-503 (p2; Rev 3), the operator will be directed to adjust Pressurizer heater demand for Banks 1 through 3 using the toggle switch on the Pressurizer Pressure Station (i.e. RCS3-PIC). Even though OP-TM-220-503 does NOT address instrument failures, control of RCS pressure is ensured more quickly and smoothly by entering and performing this procedure.

Technical Reference(s):	TQ-TM-104-220-C001 (p8-9; Rev 5) OS-24 (p13; Rev 18) OP-TM-220-503 (p2; Rev 3) OP-TM-AOP-070 (p1; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	220-GLO-11	(As available)

Question Source:	Bank #	QR-220-GLO-11-Q01	
	Modified Bank #		(Note changes or attach parent)

New

Question History:		Last NRC Exam:	None
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	7	
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate ability to operate and/or monitor the PZR heaters and sprays as they apply to the Pressurizer Pressure Control Malfunctions. This is accomplished by presenting the operator with a set of typical plant conditions, and then requiring that the operator identify the proper response, which deals with heaters and sprays, given a specific instrument failure.

The question is at the Comprehension/Analysis cognitive level because the operator must know how the Pressurizer heaters, and spray valves I s controlled, and using that knowledge, evaluate plant conditions a specific instrument failure, and determine what actions to take in response to that failure, to correctly answer the question.

What MUST be known:
1. How are the pressurizer heater banks/groups controlled? 2. What is the impact on the Pressurizer Heaters/Spray Valve of the setpoint of RCS3-PIC failing to 0 psig? 3. What actions must be taken in light of the failure of RCS3-PIC in light of the given plant conditions?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026 AA2.02	
	Importance Rating	2.9	

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss

Proposed Question: RO Question # 51

Plant conditions:

- Reactor power initially is 100%.
- Large Break LOCA occurs at RC-P-1C suction.

Based on expected plant response, which ONE (1) of the following actuation signals will result in ISOLATION of NSCC to the Reactor Building?

A. 4 psig RB Pressure ESAS.

B. 30 psig RB Pressure ESAS.

C. 500 psig RCS Pressure ESAS.

D. 1600 psig RCS Pressure ESAS.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to TQ-TM-104-642-C001 (p24-25; Rev 5) the 500 psig ESAS signal is an automatic actuation signal that is expected to occur for a major LOCA; and the operator may incorrectly believe that it is this signal that will isolate CCW.

B. **Correct.** According to TQ-TM-104-531-C001 (p15; Rev 6), the NSCCW System is isolated to the Reactor Building by three valves: (1) NS-V-15, Reactor Building component cooling supply isolation valve, (2) NS-V-35, Reactor Building component cooling inside return isolation valve and (3) NS-V-4, Reactor Building component cooling outside return isolation valve. According to TQ-TM-104-531-C001 (p38; Rev 6),

each of these valves receive an automatic close signal on 30 psig RB Isolation Signal. Additionally, according to OP-TM-244-901 (p13; Rev 3), Attachment 7.1 (p2of2), Step 7.5, the operator is directed to check that NS-V-4, 15 and 35 are closed if a 30 psig RB Isolation Signal occurs.

C. **Incorrect.** This is plausible because according to OP-TM-244-901 (p13; Rev 3), Attachment 7.1 (p1of2), Step 7.1 following a Reactor Trip or 4 psig RB Isolation Signal there are 19 valves that receive a close signal, and need to be verified. The operator may incorrectly believe that NS-V-4, 15 and 35 are among these valves.

D. **Incorrect.** This is plausible because according to TQ-TM-104-531-C001 (p38; Rev 6) NS-V-4, 15 and 35 will automatically close on a 1600 psig ESAS if the signal is coincident with low surge tank level. However, there would be no reason to think that the Surge Tank level would be low in the event of a large break LOCA.

Technical Reference(s): TQ-TM-104-531-C001 (p15; Rev 6)
OP-TM-244-901 (p13; Rev 3), Attachment 7.1 (p1-2of2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 541-GLO-5 (As available)

Question Source: Bank # IR-541-GLO-5-Q03
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2000

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Comments:

The KA is matched because the operator must demonstrate the ability to determine the cause of possible CCW loss during a specific plant event (Such as a large break LOCA).

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. NSW is isolated to the RB by three valves, NS-V-4, 15 and 35. 2. Each of these valves receive an automatic closure signal on 30 psig RB isolation Signal.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	022	AA2.04	
	Importance Rating	2.9		

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: How long PZR level can be maintained within limits

Proposed Question: RO Question # 52

Plant conditions:

- The plant is operating at 55% power.
- MU-P-1B is OOS.
- MU-P-1A is tripped.
- Letdown has been isolated.
- PZR Low Level alarm has JUST been received.
- PZR level is lowering by 1 inch every 2 minutes.

Assuming makeup cannot be restored and the current trend continues, select the approximate amount of time available for the pressurizer heaters to remain energized.

A. 2 hours

B. 2.5 hours

C. 3 hours

D. 4 hours

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** Plausible for confusion of the requirement to initiate HPI and trip the reactor when reactor >25% power and < 150 inches (tube leak procedure) to avoid emptying pzr. 2 hours divided by 2 times 1 = 60 therefore 140 inches (approximate 150).

B. **Incorrect.** Plausible for not knowing the alarm setpoint and or the level when heaters automatically de-energize.

C. **Incorrect.** Plausible for not knowing the alarm setpoint and or the level when heaters automatically de-energize.

D. **Correct.** With the alarm at 200 inches and heater cutout at 80 inches, PZR level must drop by 120 inches for heaters to de-energize. If level is dropping 1 inch every 2 minutes, then 240 minutes will pass prior to heater de-energization.

Technical Reference(s): 1101-2 Plant Setpoints (rev 72) pg 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 223-GLO-8 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate the ability to interpret the rate of change and determine how long the pressurizer will be within the limits for heaters to function.

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the current conditions, and calculate an answer.

What MUST be known:
1. The alarm setpoint on the low level for the pressurizer. 2 The level setpoint were heaters de-energize.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065	AA2.05
	Importance Rating	3.4	

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing

Proposed Question: RO Question # 53

Plant conditions:

- 100% power.
- Instrument Air pressure is 75 psig and lowering.
- The crew has implemented OP-TM-AOP-028, Loss of Instrument Air.

Assuming that the Instrument Air System continues to lower, which ONE (1) of the following identifies the MINIMUM conditions at which the reactor must be tripped?

Instrument Air pressure lowers to less than ____(1)____ on ____(2)____ Primary Instrument Air Header pressure (PI-222), and/or Secondary Instrument Air Header pressure (PI-1403).

A. (1) 70 psig
(2) EITHER

B. (1) 70 psig
(2) BOTH

C. (1) 60 psig
(2) EITHER

D. (1) 60 psig
(2) BOTH

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to OP-TM-AOP-028 (p5; Rev 5) Note prior to Step 3.6, the operator is told that when IA pressure < 70 psig, IA-P-2A and B will supply backup IA. The operator may confuse the numbers, and incorrectly believe that 70 psig is the reactor trip threshold.

B. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

C. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-AOP-028 (p3; Rev 5) Step 3.2, IAAT IA pressure is < 60 psig (PI-222 or PI-1403), then ENSURE reactor is tripped.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to OP-TM-AOP-028 (p13-15; Rev 5) Section 4.0, the response actions are conditional upon the pressure indicator that is low; and the operator may incorrectly believe that the threshold for the reactor trip is based on both parameters being low, not just one.

Technical Reference(s): OP-TM-AOP-028 (p3; Rev 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-7 and 10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

NOTE: The AOP is silent on starting a plant shutdown, focusing rather, on response efforts to regain control of IAS pressure, and to restore. The AOP does contain Reactor trip criteria, and this was used to construct the question.

The KA is matched because the operator must demonstrate the ability to determine and

interpret when to commence plant shutdown if instrument air pressure is decreasing as they apply to the Loss of Instrument Air.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. When IAS pressure is lowering, at what system pressure must the reactor be tripped? 2. When considering system pressure to apply the trip criteria, does the operator have to consider both primary and secondary pressures in the decision, or only one or the other?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	077	2.2.37	
	Importance Rating	3.6		

Generator Voltage and Electric Grid Disturbances: Equipment Control: Ability to determine operability and / or availability of safety related equipment.

Proposed Question: RO Question # 54

Plant conditions:

- 100% power.
- First Energy RCC TSO has reported that there are electrical disturbances on the grid, and that large voltage oscillations may be observed.
- The following Grid voltage is observed:

0800	231 KV
0810	229.5 KV
0820	227 KV
0830	226.5 KV
0840	225 KV
0850	223.5 KV
0900	222.5 KV
0910	222 KV
0920	224.5 KV

Assuming that all other system parameters are able to be maintained within allowable bands, which ONE (1) of the following identifies the time at which TMI is forced to initiate a 6 hour time clock to plant hot standby per Technical Specification 3.0.1?

A. 0800

B. 0850

C. 0900

D. 0910

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to 1107-11 (p18; Rev 26) the Grid Voltage Lo Alarm (LO-1) is set at 232 KV; and the operator may incorrectly believe that if the Grid Voltage drops to less than this value, TS 3.0.1 must be applied.

B. **Incorrect.** This is plausible because according to 1107-11 (p20; Rev 26) if Grid Voltage drops to <224 KV then there are specific actions called out in the procedure that must be taken; and the operator may incorrectly believe that if the Grid Voltage drops to less than this value, TS 3.0.1 must be applied.

C. **Correct.** According to 1107-11 (p19-20; Rev 26), at < 223 kV TMI is forced to initiate 6 HR time clock to plant hot standby per T.S. 3.0.1. IAAT Grid voltage < 223KV, then DECLARE both auxiliary transformers inoperable, MAKE log entry, and ENTER TS 3.0.1 action statement.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the threshold for declaring both ATs inoperable is 222KV, and it offers choice lower than the actual threshold.

Technical Reference(s): 1107-11 (p19-20; Rev 26) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 701-GLO-14 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the operator must demonstrate the ability to determine operability and / or availability of safety related equipment such as the Auxiliary Transformers.

The question is at the Memory cognitive level because the operator must recall a threshold for operability to correctly answer the question.

What MUST be known:
1. What is the threshold for Auxiliary Transformer operability on low Grid Voltage?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	025	2.1.32	
	Importance Rating	3.8		

Loss of Residual Heat Removal System: Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: RO Question # 55

Plant conditions:

- Refueling Outage is in progress.
- Decay Heat Removal Pump DH-P-1A is running.
- Decay Heat Removal Pump DH-P-1B is out of service.

The following pump parameters are observed:

DH-P-1A motor stator temperature	135°C and stable
DH-P-1A pump bearing temperature	140°F and stable
DH-P-1A motor bearing temperature	145°F and stable

Which ONE (1) of the following identifies the correct operational assessment of Decay Heat Removal Pump DH-P-1A?

A. DH-P-1A is operating with higher than normal temperatures, but within design limits.

B. DH-P-1A has exceeded its maximum allowable motor stator temperature, and must be stopped.

C. DH-P-1A has exceeded its maximum allowable pump bearing temperature, and must be stopped.

D. DH-P-1A has exceeded its maximum allowable motor bearing temperature, and must be stopped.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the other two parameters are within the design

limit allowable range, and the operator may incorrectly believe that the motor stator temperature is as well. This would be correct if the motor stator temperature were < 130°C.

B. **Correct.** According to OP-TM-212-000 (p8; Rev 15) the maximum allowable pump bearing temperature is 180°F, the maximum allowable motor bearing temperature is 190°F and the maximum allowable motor stator temperature is 130°C. The pump should not be operated if temperatures exceed these limits. Since the motor stator temperature is > than that allowed the pump must be stopped. According to TQ-TM-104-212-C001 (p46; Rev 8), regarding DHP A/B Motor Stator Temperature; 110°C temperature indicates abnormal conditions but within design limits, and 130°C temperature indicates motor winding design limitation. Regarding DHP A/B Motor Bearing Temperature; the 170°F setpoint indicates an abnormally high temperature but the temperature is within design, and the 190ºF setpoint indicates the bearing design limit is being exceeded and continued operation will result in bearing failure. Regarding DH Pump A/B Bearing Temperature; the 130°F setpoint indicates an abnormally high temperature but the temperature is within design, and the 180ºF setpoint indicates the bearing design limit is being exceeded and continued operation will result in bearing failure.

C. **Incorrect.** This is plausible because there is a maximum allowable pump bearing temperature, and the operator may incorrectly believe that it has been exceeded. This would be correct if the value were changes to >180°F.

D. **Incorrect.** This is plausible because there is a maximum allowable motor bearing temperature, and the operator may incorrectly believe that it has been exceeded. This would be correct if the value were changes to >190°F.

Technical Reference(s): OP-TM-212-000 (p8; Rev 15)
TQ-TM-104-212-C001 (p46; Rev 8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-5 and 9 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to explain and apply all system limits and precautions. This is accomplished by providing the operator with a set of DH Pump operating conditions and asking the operator to choose the correct assessment between the specific operating parameters.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What is the maximum allowable motor stator temperature for continued operation of the DH Pump? 2. What is the maximum allowable motor bearing temperature for continued operation of the DH Pump? 3. What is the maximum allowable pump bearing temperature for continued operation of the DH Pump.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	062	2.4.8	
	Importance Rating	3.8		

Loss of Nuclear Service Water: Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.

Proposed Question: RO Question # 56

Plant conditions:

- 100% power.

Event

- All Nuclear River Water pumps, NR-P-1A/B/C have tripped
- The crew enters OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, based on NS cooler outlet temperature approaching 100°F.
- OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions, have been completed.

Currently:

- A loss of 1R 480V ES Bus and 1T 480V ES Bus occurs and will not be returned to service for 24 hours.

Which ONE of the choices below states the procedures that the crew will be utilizing currently?

A. OP-TM-EOP-001, Reactor Trip, ONLY.

B. OP-TM-EOP-001, Reactor Trip, and
OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, ONLY.

C. OP-TM-EOP-001, Reactor Trip, and
OP-TM-AOP-005, River Water System Failures, ONLY.

D. OP-TM-EOP-001, Reactor Trip, and
OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, and
OP-TM-AOP-005, River Water System Failures.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.**

Although the crew will be in OP-TM-EOP-001, Reactor Trip, it will not be the only procedure because it does not address the loss of River Water cooling.
River Water Cooling was lost completely when 1R and 1T 480V ES Busses trip, deenergizing all Secondary River Water pumps. Since Nuclear River Water pumps had already tripped, entry conditions into OP-TM-AOP-005, River Water System Failures, are met and this is where the loss of River Water cooling water will be addressed.

Plausible if the examinee believes that for the given conditions, OP-TM-AOP-031 states to "GO TO" OP-TM-EOP-001.

B. **Incorrect.**

Although the crew will be in OP-TM-EOP-001, Reactor Trip, it will not be the only procedure because it does not address the loss of River Water cooling.
River Water Cooling was lost completely when 1R and 1T 480V ES Busses trip, deenergizing all Secondary River Water pumps. Since Nuclear River Water pumps had already tripped, entry conditions into OP-TM-AOP-005, River Water System Failures, are met and this is where the loss of River Water cooling water will be addressed.

Plausible if the examinee does not recognize the loss of Secondary River Water pumps and therefore the entry conditions being met for OP-TM-AOP-005.

C. **Correct.**

Definitions per OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 4.1.9 Referencing and Branching Statements:

A. **GO TO:** Leave the current step or procedure and transfer to the referenced step or procedure. The procedure which was exited is considered complete.

B. **INITIATE:** Begin the action described (steps or procedure) and continue with the other procedures in parallel.

OP-TM-AOP-031 step 3.2 states to Initiate OP-TM-EOP-001.
Additionally OP-TM-AOP-031, STEP 3.13. states:

IAAT no Nuclear River or Secondary River pumps are operating or available to be started, then GO TO AOP-005 "River Water Systems Failures"

Since this is a GO TO step, OP-TM-AOP-031 is exited upon the loss of 1R and 1T 480V ES Busses and OP-TM-AOP-005 is entered, addressing high temperatures based on a loss of River Water Systems.

D. **Incorrect.** OP-TM-AOP-031 step 3.2 states to Initiate OP-TM-EOP-001.
Additionally OP-TM-AOP-031, STEP 3.13. states:

IAAT no Nuclear River or Secondary River pumps are operating or available to be started, then GO TO AOP-005 "River Water Systems Failures"

Since this is a GO TO step, OP-TM-AOP-031 is exited upon the loss of 1R and 1T 480V ES Busses and OP-TM-AOP-005 is entered, addressing high temperatures based on a loss of River Water Systems.

Plausible if the examinee does not recognize that a GO TO condition is met which requires exiting OP-TM-AOP-031, or if the examinee does not recognize that the High Temperature conditions of OP-TM-AOP-031 will be addressed within OP-TM-AOP-005.

Technical Reference(s): OP-TM-AOP-005 (p1; Rev 9)
OP-TM-AOP-031 (p3,7; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: A05-PCO-1 and 2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of how abnormal operating procedures are used in conjunction with EOP's. In this case, the operator is in an AOP, when the AOP directs the "Initiation" of an EOP, and then directs to "Go To" another AOP. Two procedures will then be performed in parallel. The operator is required to identify which of the procedures will need to be accomplished simultaneously.

The question is at the Memory cognitive level because the operator must recall actions within OP-TM-AOP-031 and the power supplies for Secondary River Water Pumps.

What MUST be known:
1. What are the actions within AOP-031 for responding to a loss of all NR and then all NR/SR Pumps? 2. If an AOP is in progress, and the reactor trips, how should the crew address the current AOP?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	028	AK1.01	
	Importance Rating	2.8		

Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leak abnormalities

Proposed Question: RO Question # 57

Plant conditions:

- 100% power
- Pressurizer Level LT-1 is selected as the controlling channel.

Subsequently:

A leak develops on the reference leg of LT-1.
MAP G-2-5 PZR LEVEL HI/LO, alarms.

Which ONE (1) of the following describes how the Pzr level indication LT-1 compares to LT-3, AND what operator actions must be taken?

A. Indicated level on LT-1 is higher than LT-3;
The operator must place MU-V-17, Makeup Flow Control Valve, in HAND and CLOSE the valve.

B. Indicated level on LT-1 is higher than LT-3;
The operator must place MU-V-17, Makeup Flow Control Valve, in HAND and OPEN the valve.

C. Indicated level on LT-1 is lower than LT-3;
The operator must place MU-V-17, Makeup Flow Control Valve, in HAND and CLOSE the valve.

D. Indicated level on LT-1 is lower than LT-3;
The operator must place MU-V-17, Makeup Flow Control Valve, in HAND and OPEN the valve.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the instrumentation/ indication failure results in an automatic control circuit responding properly to a failed channel creating confusing competing effects. For instance, the operator may correctly conclude that the failure will result in the failed channel indicating higher than the valid channel, but take action by incorrectly responding to the failed channel which is indicating high but not actually high. If the operator were to take MU-V-17 to HAND and CLOSE the valve they would be taking an actual level that is already dangerously low, and making it go lower.

B. **Correct.** 1st part correct, 2nd part correct. According to TQ-TM-104-624-C001 (p35; Rev 2), the PZR Level instruments (0-400") RC-1- LT-1 and LT-3, are manually selected on RC-1 LR recorder on CC. The LT's use a wet reference leg such that there is a Full PZR = 0" delta-p, and an empty PZR at 400" delta-p. Consequently, a leak in the reference leg will reduce the differential pressure between the reference and the variable leg. As the differential pressure is reduced, indicated level will rise, therefore LT-1 will indicate higher than LT-3, and since it is the controlling channel, the failed channel will automatically produce an action to drive actual level in the opposite direction of the failed channel. According to TQ-TM-104-220-C001 (p23; Rev 5) Pressurizer level is maintained at 220" when the plant is operating at 100% full power. The level is sensed and applied to a level control circuit. The level control circuit when in the automatic mode will control the position of the normal makeup valve (MU-V-17). If the level drops below 220" the circuit will open the normal makeup valve until the levels returns to normal, and vice-versa. Consequently, when the reference leg leak occurs, the indicated level will start to rise, causing MU-V-217 to auto close. Since there is less makeup flow, Pressurizer level, as indicated by LT-3 will actually lower. According to OP-TM-MAP-G0205 (p1; Rev 3), the operator will be immediately called upon to make an assessment of Pzr Level indication; and if the Pzr level indication is NOT valid, place MU-V-17 is HAND and control Pzr level. The operator will be able to determine that LT-3 is the valid level indication by channel checking Safety Grade Pzr level indication LT-777. Both LT-777 and LT-3 will be showing a lower indication since the automatic Pzr level control circuit has closed MU-V-217. Because of this, according to OP-TM-MAP-G0205 (p1; Rev 3), the operator will place MU-V-17 is HAND and OPEN the valve to raise Pzr level.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the instrumentation/ indication failure results in an automatic control circuit responding properly to a failed channel creating confusing competing effects. In this instance, the operator may incorrectly conclude that the failure will result in the failed channel indicating lower than the valid channel, and then correctly conclude that actual Pzr level would need to be lowered requiring that the operator take MU-V-17 to HAND and CLOSE.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the instrumentation/ indication failure results in an automatic control circuit responding properly to a failed channel creating confusing competing effects. In this instance, the operator may incorrectly conclude that the failure will result in the failed channel indicating lower than the valid channel, and then conclude that Pzr level needed to be raised requiring that the operator take MU-V-17 to HAND and OPEN.

Technical Reference(s):	TQ-TM-104-624-C001 (p35; Rev 2) TQ-TM-104-220-C001 (p23; Rev 5)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	624-GLO-5, 220-GLO-5	(As available)

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

Question History:	Last NRC Exam:	N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the operational implications of PZR reference leak abnormalities as they apply to Pressurizer Level Control Malfunctions.

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate understanding of how the Pressurizer Level detects level, and then determine how a failure of a reference leg affects actual pressurizer level to correctly answer the question.

What MUST be known:
1. How will the pressurizer indicate when there is minimum differential pressure between the reference and the variable leg of the Pressurizer level detector? 2. How will a leak in the reference leg of the Pressurizer level detector affect indicated level? 3. How does the automatic control of Pressurizer level work? 4. What action is necessary when a failed controlling Pzr Level channeling has resulted in a Hi/Lo alarm?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	A07	AK2.2	
	Importance Rating	3.3		

Knowledge of the interrelations between the (Flooding) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: RO Question # 58

Plant conditions:

- 100% power.
- The Susquehanna River is flooding, and is currently at 294 feet and rising.
- OP-TM-AOP-002, Flood, has been implemented.

Which ONE (1) of the following identifies the MINIMUM (lowest) ACTUAL level at which a plant shutdown and cooldown to Cold Shutdown will be required to be initiated?

A. 300 feet

B. 302 feet

C. 303 feet

D. 304 feet

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-AOP-002 (p15; Rev 4), Section 4 will be initiated at a river elevation of 300 feet. Section 4.0 indicates that the unit status is such that the flood level of 300' or greater is predicted to occur within 36 hours and that necessary flood protection actions that affect equipment operability and install flood gates on designated building access doors will begin. The operator may incorrectly believe that this section requires a plant shutdown and cooldown.

B. **Correct.** According to OP-TM-AOP-002 (p21; Rev 4) Step 4.22, IAAT the river level

exceeds 302', the operator is directed to Section 5.0. According to OP-TM-AOP-002 (p27; Rev 4), Section 5.0 states that this section is entered when River level at the ISPH is approaching or is above 302' elevation, and that under these conditions reactor shutdown and cooldown are required. The operator will be directed by Step 5.3 to initiate a plant shutdown IAW 1102-10, "Plant Shutdown." Then, the operator will be directed by Step 5.4 to initiate a cooldown to Cold Shutdown IAW 1102-11, "Plant Cooldown," after the plant reaches Hot Standby.

C. **Incorrect.** This is plausible because according to OP-TM-AOP-002 (p17; Rev 4) Note prior to Step 4.6, flood barriers must be in place before river water elevation reaches 303 feet. Additionally, according to OP-TM-AOP-002 (p31; Rev 4) Step 5.11, IAAT the river elevation exceeds 303 feet, the operator is directed to station two operators in the Diesel generator Building, the Intermediate Building, the ISPH, and at Unit 2. The operator may incorrectly believe that this is the threshold at which a plant shutdown and cooldown are required.

D. **Incorrect.** This is plausible because according to OP-TM-AOP-002 (29; Rev 4) Steps 5.6 and 5.7, the 304 foot level is a threshold for which a specific set of actions is provided in AOP-002; and the operator may incorrectly believe that the shutdown/cooldown is required at this level. Additionally, according to Technical Specification 3.14.2 (p3-60; Amendment 157), If the river stage reaches elevation 302 feet at the River Water Intake Structure, the unit will be brought to the hot standby condition. The operator may know the Technical Specification, and incorrectly believe that the requirement to initiate a Plant Cooldown is at a higher elevation.

Technical Reference(s): OP-TM-AOP-002 (p15, 17, 21, 27 29 and 31; Rev 4) Technical Specification 3.14.2 (p3-60; Amendment 157) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP DBIG PCO-3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between Flooding and the facility's heat removal systems, and relations between the proper operation of these systems to the operation of the facility. This is accomplished by requiring the operator to identify at what river level threshold these heat removal systems will be required to be employed.

NOTE: According to Technical Specification 3.14.2, the design flood corresponds to an elevation of approximately 303 feet at the River Water Intake Structure. The dike elevation at the intake structure is 305 feet. The minimum freeboard is at the downstream end of the plant site where the dike elevation is 304 feet providing a freeboard of approximately one foot. Adequate freeboard is provided to protect the plant site from flooding due to wave action during the design flood. Placing the unit in hot standby when the river stage reaches 302 feet elevation provides an additional margin of conservatism by assuring that adequate freeboard exists during operation of the unit. The OP-TM-AOP-002 adds additional margin by requiring not only placing the unit in Hot Standby, but cooldown of the plant as well.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What is the Technical Specification elevated river level requiring plant shutdown? 2. AOP-002 requires that the plant be unconditionally brought to Hot Standby and then Cold Shutdown when the river level rises to 302 feet.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A06 AK3.3	
	Importance Rating	4.2	

Knowledge of the reasons for the following responses as they apply to the (Shutdown Outside Control Room) : Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Proposed Question: RO Question # 59

Plant Conditions

- Control tower has been destroyed.
- 4 hours since event.
- OTSG Pressures are 60 psig and lowering.

Which ONE (1) of the following identifies how OTSG levels will be maintained?

A. CO-T-1A/B via EF-P-1.

B. The river via RR-P-1A or B.

C. Fire Service Water via FS-P-15.

D. Million Gallon Tank via EF-P-2A or B.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because for the first two hours of the event, this is the expected source of EFW flow to the OTSGs. The operator may incorrectly believe that this method is used longer than four hours into the event.

B. **Incorrect.** This is plausible because flow can be delivered to the OTSGs using this means if 4160V Bus is available. However, according to OP-TM-AOP-0091 (p2; Rev 1), it is assumed as a condition of entry that there has been a total loss of AC power when this procedure is used. Consequently, the procedure addresses EFW flow strategies accordingly, and does not address the use of the RR Pumps.

C. **Correct.** According to OP-TM-AOP-009 (p5-7; Rev 5) Steps 3.11 through 3.15, the operator will be directed to wait 2 hours after EFW has been established (which was established at Step 3.2, almost immediately upon entry into AOP-009), and throttle EFW flow to 400 gpm. Step 3.6 of this procedure has previously established 600 gpm of EFW flow to the OTSGs. Then, the operator will be directed to align Fire Pump FS-P-15 (portable pump) for operation, and once OTSG pressure is < 250 psig, start FS-P-15, and provide 200 gpm of EFW flow using the fire pump. After this, the operator will be directed to stop EF-P-1.

D. **Incorrect.** This is plausible because flow can be delivered to the OTSGs using this means if 4160V Bus is available. However, according to OP-TM-AOP-0091 (p2; Rev 1), it is assumed as a condition of entry that there has been a total loss of AC power when this procedure is used. Consequently, the procedure addresses EFW flow strategies accordingly, and does not address the use of the Motor Driven EFW Pumps.

Technical Reference(s): OP-TM-AOP-009 (p5-7; Rev 5)
OP-TM-AOP-0091 (p2; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: A09-PCO-4 (As available)

Question Source: Bank # QR-A09-PCO4-Q02
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content: 55.41 5
55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:
The KA is matched because the operator must demonstrate Knowledge of the reasons for the manipulation of controls required to obtain desired operating results during abnormal, and emergency situations as they apply to Shutdown Outside Control Room. This is accomplished by presenting the operator with a set of conditions that would require the use of AOP-009, 4 hours into the event, and requiring the operator to identify the method used at this point to feed the OTSGs. By identifying the correct answer, the operator has demonstrated knowledge of the reason for control manipulation. If the operator does not know that EF-P-1 could be damaged if operated under the stated conditions, this distractor might be chosen. Likewise, if the operator chooses one of the distractors requiring the use of the 4160V Bus, then the operator does not have knowledge of the assumptions made in developing the procedure.

The question is at the Comprehension/Analysis cognitive level because the operator must know bits of information regarding the strategy for feeding the OTSGs during a loss of control tower event, and then apply that information to a specific time sequence within the event to correctly answer the question.

What MUST be known:
1. What is the initial method used to fill the OTSGs on a loss of Control Tower? 2. When does this method need to change? 3. In the given conditions, what method will be used to provide EFW flow to the OTSGs?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	001	AA1.07	
	Importance Rating	3.3		

Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal: RPI

Proposed Question: RO Question # 60

Plant conditions:

- 80% power.
- A test of Group 7 Control Rods is in progress per OP-TM-622-201, Control Rod Movement.
- Group 7 Control Rods are initially at 60%.

Event:

- The operator inserts the Group 7 Control Rods for 12 seconds.
- When the operator withdrawal's the rods to 60% and releases the handle the operator observes that the API Bars for Group 7 Control Rods continue to move in an upward direction on the Rod Position Indication Panel.

Which ONE (1) of the following identifies an indication on the Rod Position Panel that would validate that a Continuous Rod Withdrawal is occurring?

A. The Group 7 rod position bars turn dark green.

B. The red blocks above the Group 7 rod position bars light.

C. The background of the Group 7 average indicator turns yellow.

D. There is an orange triangle on the right side of the panel pointing upwards.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** This is plausible because according to OP-TM-622-000 Rev 5 Attachment 7.5 Pg 6 of 7, In Limits are indicated by turning the rod position bars dark-green. Under normal operation the operator should not see the rod position bars dark green, and may

be unfamiliar with this indication.

B. **Incorrect.** This is plausible because according to OP-TM-622-000 Rev 5 Attachment 7.5 Pg 6 of 7, Out Limits are indicated by displaying a red block at the top of the rod position bars. However, this will not occur until the Group 7 Control Rods are at the top of the core.

C. **Incorrect.** This is plausible because according to OP-TM-622-000 Rev 5 Attachment 7.5 Pg 6 of 7, the API Group Average is displayed above the corresponding group's control rod display as a numerical value to one decimal place; and the background of the group 7 average indicator turns yellow when the group 7 average is greater than 97%. The operator may incorrectly believe that the background turns yellow at a lower value, and is indicative of a CRWA.

D. **Correct.** According to OP-TM-622-000 Rev 5 Attachment 7.5 Pg 6 of 7, the right display indicates rod insertion and withdrawal using an orange triangle pointing in the direction of rod movement. When no rod motion is commanded, an empty rectangle is displayed. Since the operator released the withdrawal handle, there is no rod motion commanded, and an empty rectangle should be displayed. However, the orange triangle pointing upward, toward the top of the core is indicative, that a Continuous Rod Withdrawal Accident is occurring.

Technical Reference(s): OP-TM-622-000 (p74; Rev 5)
OP-TM-622-201 (p5; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO-5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

The KA is matched because the operator must demonstrate the ability to operate and / or monitor the Rod Position Indication Panel as it applies to the Continuous Rod Withdrawal.

The question is at the Memory cognitive level because the operator must recall bits of information to answer the question correctly.

What MUST be known:
1. What is the meaning Control Rod position bars turning dark green on the RPI Panel right display? 2. What is the meaning of the orange up arrow in the RPI Panel right display? 3. What is the meaning of the red block light above the Control Rod Position Bars on the RPI Panel, and the significance of it being Lit? 4. Under what conditions would the API Average Indicator background turn yellow?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051	AA2.02
	Importance Rating	3.9	

Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

Proposed Question: RO Question # 61

Plant conditions:

- A Reactor Startup is in progress with Reactor Power at 40%.
- Circulating Water Pump CW-P-1B is removed from service.

The following events occur:

- Annunciator N-1-6, MN COND VACUUM LO, is in Alarm.
- ARO reports Condenser backpressure is 5.6" HgA.
- The crew manually trips the Main Turbine IAW MAP N-1-6.
- Vacuum drops to 22" HgA and Atmospheric Dump Valves MS-V-4A/4B assume pressure control from MS-V-3A thru F.

In accordance with MAP N-1-6, which ONE (1) of the following actions is required to be taken NEXT by the crew?

A. Trip the Reactor; perform EOP-001.

B. Restore normal control of MS-V-4A/4B.

C. Verify all available Circ Water Pumps are operating.

D. Perform OP-TM-301-151, Main Turbine Generator Operating Mode to Standby Mode.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-MAP-N0106 (p1; Rev 8) MAP Step 4.1, the operator will be directed to IAAT Condenser Backpressure greater than Alarm Setpoint, then if

Generator load is less than 272 MWe, then TRIP the Turbine. It is given in the initial conditions that the Turbine has been tripped. Secondly, the operator is directed to reduce load IAW 1102-4, "Power Operation". Step 4.2 provides the operator with additional conditional information. The operator is then immediately directed to IAAT OTSG pressure is being controlled using ADVs, then ensure that the Reactor is tripped. Since the stated conditions indicate that this is the case, the reactor must be tripped and EOP-001 is entered.

B. **Incorrect.** This is plausible because according to OP-TM-MAP-N0106 (p1; Rev 8) MAP Step 4.5, normal control of MS-V-4A / 4B is performed if pressure control is transferred to ADV's and Condenser vacuum is > 25" Hg VAC. The operator may not know the requirement to trip, or the threshold at which normal steam pressure control is established.

C. **Incorrect.** This is plausible because according to OP-TM-MAP-N0106 (p1; Rev 8) MAP Step 4.3, the operator is directed to check all Circ Water Pumps operating if there is low Circ Water flow or high Circ Water Temperature. The operator may not know the requirement to trip, and may believe that this is the correct choice. However, the procedure states to check the Circ Water Pumps operating not available.

D. **Incorrect.** This is plausible because according to MAP-K0101 (p2; Rev 5), Step 4.7, this would be an expected action in response to the Turbine Trip. The operator may incorrectly believe that a reactor trip is not required.

Technical Reference(s): OP-TM-MAP-N0106 (p1; Rev 8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 421-GLO-10 (As available)

Question Source: Bank # IR-421-GLO-10-Q02

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: None

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to determine and interpret conditions requiring reactor trip as it applies to the Loss of Condenser Vacuum.

The question is at the Comprehension/Analysis cognitive level because the operator must know bits of information and then apply that to a set of plant conditions to correctly answer the question.

What MUST be known:
1. The reactor will not automatically trip when the turbine is tripped when reactor power is < 45%. 2. The reactor is required to be manually tripped in a low condenser vacuum situation when the ADVs are controlling OTSG pressures. 3. The plant conditions require a manual reactor trip.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	061	2.4.8	
	Importance Rating	3.8		

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.

Proposed Question: RO Question # 62

With the plant at 100% power the following events occur:

- RM-G-19, Reactor Coolant Pump Seal Return, goes into ALARM.
- The crew closes MU-V-25, Seal Water Return RB Inner Isolation Valve, and MU-V-26, Seal Water Return RB Outer Isolation Valve.
- The Reactor Coolant Drain Tank level rises to 100%.
- The crew enters OP-TM-AOP-050, "Reactor Coolant Leakage."

Subsequently:

- The reactor automatically trips.
- RCS parameters stabilize at normal post-trip operating pressure and temperature.

Which ONE (1) of the following identifies the manner in which the procedures must be used?

A. OP-TM-AOP-050 will be exited.
The crew will perform all of the actions of OP-TM-EOP-001, Reactor Trip.

B. OP-TM-AOP-050 will be interrupted to perform all of the actions of OP-TM-EOP-001, Reactor Trip;
Once complete, the Control Room Supervisor will continue with OP-TM-AOP-050.

C. OP-TM-AOP-050 will be interrupted to perform OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions and the initial Symptom Check;
Once complete, the Control Room Supervisor determines the sequence of action between the parallel procedures.

D. OP-TM-AOP-050 will be performed in parallel with OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions and the initial Symptom Check;
Once complete, the Control Room Supervisor determines the sequence of action between the parallel procedures.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that the AOP is always suspended, or exited by the higher priority EOP.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the AOP is performed after the completion of the higher priority EOP.

C. **Correct.** According to OS-24 (p8; Rev 19) performing Parallel Procedures any other procedure actions should be interrupted to perform Reactor Trip Immediate Manual Actions and the initial Symptom Check. Once Immediate Manual Actions and the initial symptom check have been accomplished, the Control Room Supervisor determines the sequence of action between parallel procedures.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that the rule for parallel path procedure implementation includes parallel performance with the EOP-001 IMAs and VSSV.

Technical Reference(s): OS-24 (p8; Rev 19) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP001-PCO-5 and 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate Knowledge of how abnormal operating procedures are used in conjunction with EOP's in a setting in which an Area Radiation Monitor has alarmed.

The question is at the Memory cognitive level because the operator must recall the rule of implementation for AOPs and Reactor Trip IMAs/VSSV, in order to answer the question correctly.

What MUST be known:
1. If an AOP is in progress, and the reactor trips, how should the crew address the current AOP?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	060	EA2.05	
	Importance Rating	3.7		

Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: That the automatic safety actions have occurred as a result of a high ARM system signal

Proposed Question: RO Question # 63

Plant conditions:

- 100% power.
- RM-G-20, Reactor Coolant Drain Tank, goes into HI ALARM.

Which ONE (1) of the following identifies the Waste Gas (WDG) valve(s) that will automatically CLOSE?

A. WDG-V-2, RDCT Vent Valve, ONLY.

B. WDG-V-2, RDCT Vent Valve, and WDG-V-3, RB Vent Header Isolation Valve, ONLY.

C. WDG-V-3, RB Vent Header Isolation Valve, and WDG-V-4, RB Vent Header Outside Isolation Valve, ONLY.

D. WDG-V-2, RDCT Vent Valve, WDG-V-3, RB Vent Header Isolation Valve, and WDG-V-4, RB Vent Header Outside Isolation Valve.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may not know about the WDG-V3 and 4 closure, and according to TQ-TM-104-231-C001 (p17; Rev 2), WD-V-2 prior to a recent modification had an automatic closure on high RDCT pressure. The operator may incorrectly believe that this is the only valve that closes on Hi Area Radiation.

B. **Incorrect.** According to TQ-TM-104-231-C001 (p26; Rev 4), both WDG-V-3 and WDG-V-2 are ES powered (4160VAC Bus D/E) valves, while WDG-V-4, is a SOV, powered

from relay panel XCL, which is 125 VDC - BOP. Because of this, the operator may incorrectly believe that it is WDG-V-2 and 3 that auto CLOSE, and not WDG-V-3 and 4.

C. **Correct.** According to TQ-TM-104-661-C001 (p19; Rev 4), RM-G-20, RCDT Discharge monitor, is located in the Auxiliary Building elevation 305' next to the Decay Heat Closed Cooling Water pump area and monitors the reactor coolant drain tank line. This monitor was installed as a restart modification task RM-5(b) and provides interlock signals to close Reactor Building isolation valves WDL-V-303, 304, WDG-V-3, 4 to isolate the line should the radiation level exceed the setpoint. According to TQ-TM-104-661-C001 (p52; Rev 4), RM-G-20 shuts WDG-V-3/4 and WDL-V-303/304 (isolates gaseous and liquid [respectively] discharge from the RCDT). Additionally, according to OP-TM-MAP-C0101 (p38; Rev 1), WDG-V-3 and 4 will automatically CLOSE when RM-G-20 goes into HI ALARM.

D. **Incorrect.** This is plausible because each of these valves will isolate the RDCT Vent Header, and this would be an appropriate automatic action when hi radiation is sensed within the area.

Technical Reference(s):	TQ-TM-104-661-C001 (p19 and 52; Rev 4) OP-TM-MAP-C0101 (p38; Rev 1) TQ-TM-104-231-C001 (p17 and 26; Rev 2)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

The KA is matched because the operator must demonstrate ability to determine and interpret that the automatic safety actions have occurred as a result of a high ARM system signal as they apply to the Accidental Gaseous Radwaste Release.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. What automatic actions occur when RM-G-20 goes into high alarm?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	E09	EK3.4	
	Importance Rating	3.8		

Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Cooldown) RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Proposed Question: RO Question # 64

Plant conditions:

- A LOCA has occurred.
- All RCPs are OFF.
- The crew has entered Section 4.0, Inadequate RCS Cooldown, of OP-TM-EOP-006, LOCA Cooldown.
- The crew is in the process of raising OTSG levels.

Which ONE of the following identifies the OTSG level that is required to be maintained per OP-TM-EOP-006, the preferred source of feedwater, AND the reason that this source is preferred per the EOP-006 basis document?

A. 50% in the Operating Range;
EFW; AND
This is because EFW has a higher injection point on the OTSG then MFW does, and this is needed to promote Natural Circulation flow.

B. 50% in the Operating Range;
MFW; AND
This is because MFW will have a less adverse effect on the Tube-to-Shell Differential Temperature (TSDT).

C. 75-85% in the Operating Range;
EFW; AND
This is because EFW has a higher injection point on the OTSG then MFW does, and this is needed to promote Natural Circulation flow.

D. 75-85% in the Operating Range;
MFW; AND
This is because MFW will have a less adverse effect on the Tube-to-Shell Differential

Temperature (TSDT).

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to OP-TM-EOP-010, Rule 4 (p9; Rev 12), 50% OTSG operating level corresponds to the normal post-trip OTSG level.

B. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

C. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-006 (p21; Rev 8) Step 4.6, the operator will be directed to raise OTSG level to 75 to 85% with EFW. If the operator cannot do this, the RNO will direct the operator to raise OTSG level to 75 to 85% with MFW. So, the OTSG level band is 75-85%, and the preferred source of water is EFW. According to OP-TM-EOP-0061 (p17; Rev 4), The step intent is to increase the relative strength of the OTSG heat sink. If steam voids in the RCS hot legs have blocked natural circulation flow, establishing OTSG levels at the loss of subcooled margin setpoint will provide cooling and subsequent collapse of the steam void, allow refill of the hot leg and reestablish conditions favorable for natural circulation. EFW is the preferred source due to injection point in the OTSGs, however, MFW is acceptable if EFW is not available.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to OP-TM-EOP-010 (p25; Rev 12) when feeding a dry or depressurized OTSG, the operator is directed to feed the OTSG with MFW as the preferred source over EFW. While doing so the operator is told that it is anticipated that Tube to Shell delta T limits will be exceeded. The operator may confuse the concepts.

Technical Reference(s): OP-TM-EOP-006 (p21; Rev 8)
OP-TM-EOP-0061 (p17; Rev 4)
OP-TM-EOP-010, Rule 4 (p9 and 25; Rev 12) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP006-PCO-3 and 4 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of the reasons for RO functions within the control room team, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated as they apply to the Natural Circulation Cooldown. This is accomplished by presenting a set of conditions that require the promotion of natural circulation, and requiring the operator to identify the OTSG level band and preferred source of feedwater flow, and the reason for the preferred source of feedwater flow

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate an understanding of why an action is taken in the manner in which it is to successfully answer the question.

What MUST be known:
1. What is the OTSG level band in an NC Cooldown? 2. What is the preferred source of feedwater flow for raising the level of the OTSG in the given situation? 3. Why is EFW the preferred source of feedwater flow in the given situation?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A04	AA1.3
	Importance Rating	3.5	

Ability to operate and / or monitor the following as they apply to the (Turbine Trip): Desired operating results during abnormal and emergency situations.

Proposed Question: RO Question # 65

Plant conditions:

- Reactor Power is 25% and steady.
- The Main Turbine has just been synchronized onto the electrical grid.
- Main Turbine exhaust Hood temperature starts to rise.

Which ONE of the following identifies:

(1) The Exhaust Hood Temperature at which the operator must trip the Turbine, AND
(2) Once tripped, the first action to be taken if Turbine Stop Valve SV-1 and Turbine Control Valve CV-1 do **NOT** close?

A. (1) 175°F;
(2) Trip the Reactor.

B. (1) 175°F;
(2) Open Bypass Valve EHC-FV-1.

C. (1) 225°F;
(2) Trip the Reactor.

D. (1) 225°F;
(2) Open Bypass Valve EHC-FV-1.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to OP-

TM-301-000 (p3; Rev 16), 175F is the threshold for Exhaust Hood Temperature that causes the hi temperature alarm to come in. The operator may incorrectly believe that a Turbine Trip is required at this point.

B. **Incorrect.** 1st part wrong, 2nd part wrong. See A and D.

C. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-301-000 (p3; Rev 16), after initial synchronizing of the machine and while at low loads, exhaust hood temperature could go above the 125°F operating range. If so, the load should be raised slowly until the temperature falls below 125°F. High exhaust hood temperature alarms at 175°F. The turbine should be manually tripped if hood temperatures cannot be maintained less than 225°F. According to OP-TM-MAP-K0101 (p2; Rev 5), the Reactor will not trip, unless Reactor power level >45 percent. Additionally, according to Step 4.4, if the MS-V-1s are OPEN, and if Turbine stop valve or control valve closure has not isolated flow to HP turbine, then TRIP the Reactor.

D. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to OP-TM-EOP-001 (3; Rev 10) Step 2.4 RNO, the operator is directed to PLACE both EHC-P-1A and EHC-P-1B in Pull-To-Lock, and PLACE both EHC-P-1A and EHC-P-1B in Pull-To-Lock. The operator may confuse the EOP-001 action with the action required by OP-TM-MAP-K0101.

Technical Reference(s): OP-TM-301-000 (p3; Rev 16)
OP-TM-MAP-K0101 (p2; Rev 5)
OP-TM-EOP-001 (3; Rev 10) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 301-GLO-10 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate ability to operate and / or monitor for desired operating results during abnormal and emergency situations as they apply to the Turbine Trip.

The question is at the Memory cognitive level because the operator must simply recall bits of information to correctly answer the question.

What MUST be known:
1. What is the threshold for tripping the Turbine on high Turbine Exhaust Hood Temperature. 2. What action must be taken when a Turbine Stop Valve does not fully close on a Turbine Trip when the reactor is still operating.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.45	
	Importance Rating	4.3		

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indicator.

Proposed Question: RO Question # 66

Plant conditions:

- The plant is shutdown in Draindown Mode.
- The Reactor Vessel Level is being maintained at 61±2 inches.

The following is observed:

- Draindown Reactor Vessel Level Instruments RC-LT-1037 is 62 inches.

Which ONE (1) of the following identifies an indication that would validate the accuracy of this instrument reading?

A. Pressurizer Level as indicated on RC-LI-777 is 59 inches.

B. Draindown Reactor Vessel Level Center Tygons are 64 inches.

C. Draindown Reactor Vessel Level Instruments RC-LT-1138 is 59 inches.

D. Pressurizer Level as indicated on PPC Point C4231, Cold Pressurizer Level, is 64 inches.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may incorrectly believe that this instrument can be used under the stated conditions, that this level is within the allowable range (which can be confusing at 8±4 inches below draindown level indicators), and that it is the most accurate of the Pressurizer Level indications.

B. **Correct.** According to 1103-11 (p46; Rev 69A), when monitoring the water inventory in Reactor Vessel or in areas of Cold legs or Hotlegs which will drain by gravity to the

reactor vessel, water level is measured in inches above 314' 0" (i.e. Cold leg centerline). The Reactor Vessel water level indications are RC-LI-1138, RC-LI-1037 and tygon tubing used for manometer level indication at RC-V-1320 and RC-V-1323 (hot leg /cold leg centerline). Additionally, the Pressurizer level can be monitored as well. When the pressurizer is cold and is near atmospheric pressure, the most accurate indication of pressurizer level is obtained by using PPC point C4231 Cold Pressurizer Level. According to 1103-11 (p26; Rev 69A) If the RCS is to be drained to < 100" Pressurizer Level, then when RCS Temperature < 250°F, then INSTALL "Tygon" tubing for use as level indicators at the following locations: At RC-V-1323 (3/4" ID clear 50' long) [Cold Leg Centerline- RC-P-1A] and at RC-V-1320 (3/4" ID clear 50' long) [Cold Leg Centerline- RC-P-1B]. Since the Pressurizer Level is drained to < 100 inches, these are in service. According to 1103-11 (p15; Rev 69A), the operator is directed to VERIFY two or more RCS water level indicators are operable. If Pzr level < 67, then these indicators must be Reactor Vessel water Level Indicators. If Pzr level > 67" then Pressurizer level indicators may be used as well. Since water level is less than 67 inches, the pressurizer level cannot be used. Therefore, the only indications that may be used as the diverse indication for a Draindown level instrument are the other instrument (i.e. RC-LT-1138 TEMP) and Cold leg centerline tygons. According to 1103-11 (p46; Rev 69A) in order to avoid initiation of Attachment 2, RCS Water Level Indication Quality Questionable, RC-LT-1137 TEMP and RC-LT-1138 TEMP must differ by no more than 2 inches, and these instruments must not differ by more than 2 inches from the Center Tygons. Therefore, since RC-LT-1138 TEMP is 59 inches, it does not corroborate RC-LT-1037-TEMP. On the other hand, since the Draindown Reactor Vessel Level Center Tygons are 64 inches, and within 2 inches, these instruments do corroborate this reading.

C. **Incorrect.** This is plausible because the operator may incorrectly believe that the backup instrument is within the allowable band.

D. **Incorrect.** This is plausible because the operator may incorrectly believe that this instrument can be used under the stated conditions. It is within range, and the most accurate of the Pressurizer Level indications during draindown.

Technical Reference(s): 1103-11 (p15, 25-27, 45-46; Rev 69A) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

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

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate ability to identify and interpret diverse indications to validate the response of another indicator. In this case the operator is presented with a set of plant conditions, including a Reactor Vessel Water Level to maintain, and then must choose which instrument/reading can be used to validate a specific level instrument.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then apply them to a specific set of plant conditions to correctly answer the question.

What MUST be known:
1. When are the Cold Leg Centerline Tygons placed in service, and available for use as a backup means of RV Level indication? 2. When is the Pressurizer Level able to be used to back up the RV Level indication? 3. When is a diverse indication considered to be questionable? 4. In the specific plant conditions, which instruments are able to serve as backup indication?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.25	
	Importance Rating	3.9		

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 67

Plant Conditions:

- Plant operating at 100% power.
- Main Generator H2 leak has occurred.
- Generator machine gas pressure is 45 psig and H2 leak secured.
- Plant MVAR loading is +120 MVAR.
- Auto makeup of hydrogen has been secured.

Which ONE (1) of the following identifies the approximate MAXIMUM generator megawatt output allowed for these plant conditions?

A. 720 MW

B. 820 MW

C. 920 MW

D. 1020 MW

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because the operator may not know how to use the curve, or use it incorrectly.

B. **Incorrect.** This is plausible because the operator may not know how to use the curve, or use it incorrectly. If the operator uses the 30 psig Curve rather than the 45 psig curve, this would be the correct answer.

C. **Correct.** The operator will use OP-TM-301-472 (p5; Rev 5) Attachment 7.1 (Provided) to determine the maximum generator megawatt output. When this is done the operator will locate the intersection of the 120 MVAR horizontal line with the 45 psig curve and determine Y Axis that the maximum generator output is 920 MW.

D. **Incorrect.** This is plausible because the operator may not know how to use the curve, or use it incorrectly. If the operator uses the 60 psig Curve rather than the 45 psig curve, this would be the correct answer.

Technical Reference(s): OP-TM-301-472 (p5; Rev 5) Attachment 7.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OP-TM-301-472, ATT 7.1

Learning Objective: 711-GLO-10 (As available)

Question Source: Bank #

Modified Bank # QR-711-GLO-10-Q01 (Note changes or attach parent)

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to interpret reference materials, such as graphs, curves, tables, etc. This is accomplished by providing the operator with a set of plant conditions, and Attachment 7.1 of OP-TM-301-472, and requiring that the operator determine the approximate maximum generator output value.

The question is at the Comprehension/Analysis cognitive level because the operator must use bits of information, and apply that information to a curve to correctly answer the question.

What MUST be known:
1. How to locate the MVAR loading on Attachment 7.1. 2. Which curve to use on Attachment 7.1. 3. The maximum generator output can be determined by reading the Y Axis at the intersection of the location of the MVAR loading and 45 psig hydrogen curve.

The question is significantly modified because two of the fill-in choices have been reversed resulting in a new correct answer. According to NUREG-1021, ES-401Section D.2.f, paragraph 1, bullet 4; to be considered a significantly modified question, at least one pertinent condition in the stem and at least one distractor must be changed from the original bank question. Changing the conditions in the stem such that one of the three distractors in the original question becomes the correct answer would also be considered a significant modification. This question has had the MVAR loading as well as the hydrogen pressure change from the parent question. Additionally, 2 of the 4 potential answers have been changed, and the correct answer is different from that of the parent.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.1	
	Importance Rating	4.5	

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: RO Question # 68

Initial Conditions:

- Reactor startup is in progress IAW OP 1103-8, Approach to Criticality.
- IAW OP-TM-300-403, "Estimated Critical Rod Position" Attachment 7.1:
    - o Estimated Critical Rod Position is 50% withdrawn on CRG 6.
    - o Minimum Rod Withdrawal Limit is 95 % rod index.
    - o Maximum Rod Withdrawal Limit is 225 % rod index.
- Initial count rate is 4×10^1 cps on NI-11 and NI-12.

Current Conditions:

- Current Rod Position is 25% on CRG 7.
- Current count rate is stable at 6×10^2 cps on NI-11 and NI-12.

Which ONE (1) of the following identifies the next action to be taken?

A. Begin emergency boration to achieve 1% dk/k shutdown margin.

B. Continue to withdraw Control Rod Group 7 and continue the reactor startup.

C. Insert Rods in sequence until Group 7 Rods and Group 6 Rods ONLY, are fully inserted.

D. Insert Rods in sequence until Group 7 Rods, Group 6 Rods, and Group 5 Rods are fully inserted.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because according to 1103-8 (p3; Rev 53) Step 2.5, under certain conditions in 1103-8 boration to achieve 1% dk/k shutdown margin is an acceptable action.

B. **Incorrect.** This is plausible because the operator may incorrectly believe that the ECP is still valid.

C. **Correct.** According to 1103-8 (p9; Rev 53), If the ECP was missed or ECB/ECP becomes INVALID, then INSERT control rods in sequence, until the rod group which was being withdrawn is fully inserted and one additional group is fully inserted. Since Group 7 rods are being withdrawn, the operator must insert these rods, and then the Group 6 Rods as well.

D. **Incorrect.** This is plausible because the operator may correctly believe that the ECP is invalid, but incorrectly conclude that at least two Groups of Rods above and beyond those currently being withdrawn must be inserted.

Technical Reference(s): 1103-8 (p3 and 9; Rev 53) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-003-PCO-4 (As available)

Question Source: Bank # IR-GOP-003-PCO-4-Q04

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: None

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load

changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate the ability to perform pre-startup procedures for the facility (i.e. Approach to Criticality), including operating those controls associated with plant equipment that could affect reactivity (i.e. Insert Rods).

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information to correctly answer the question, and then apply this information to the current set of conditions to correctly answer the question.

What MUST be known:
1. What are the requirements for a valid ECP? 2. Under the stated conditions, is the ECP valid? 3. What action is taken when the ECP maximum has been exceeded?

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2	2.2.18
	Importance Rating	2.6	

Equipment Control: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Proposed Question: RO Question # 69

Plant Conditions:

- The plant has been shut down for 1 hour.
- Hot Shut Down conditions have <u>NOT</u> been confirmed.
- NI-11/NI-11A source range instruments are OOS.
- WDL-LT-804 Normal Reactor Building Sump level channel is OOS.
- RC-V-42 Reactor Vessel Head Vent is OOS.
- RC-V-40A and RC-V-40B Reactor Coolant System A & B High Point Vents are OOS.
- NS-P-1B is OOS.

Which ONE (1) of the following must be protected?

A. VBB Vital Bus "B".

B. 480V AC ES Bus 1P and 1S.

C. RM-A-2, Reactor Building Radiation Monitor.

D. RC-V-28 and RC-V-44, Pressurizer Steam Space Vents.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-AA-108-117 (p4; Rev 2), when SSCs are planned to or become unavailable, then PROTECT redundant equipment if plant configuration is such that redundant equipment unavailability or manipulation would cause (1) an overall online or outage risk assessment change to red risk, (2) a loss of generation capability of > 20 MWe, or (3) an entry into Tech Spec 3.0.1or a shutdown Tech Spec LCO of 12 hrs or less (i.e. be in hot shutdown in 12 hrs or less). According to Technical Specification 3.5.1 (p3-27; Amendment 189) the reactor shall not be in a startup mode

or in a critical state unless the requirements of Table 3.5-1, Column "A" and "B" are met, except as provided in Table 3.5-1, Column "C". Specification 3.0.1 applies. According to Table 3.5-1 A.4 (p3-29; Amendment 247) there is a minimum of one Source Range Instrument Channel required to be OPERABLE when the reactor in a startup mode or in a critical state. Since Hot Shutdown conditions have not been verified, then one SR instrument is required to be OPERABLE. If it is not, according to Table 3.5.1 A.4 ACTION (p3-30; Amendment 189), restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours. Both this criterion, and the fact that Technical Specification 3.0.1 will force Source Range Channel N-12 to be protected. According to OP-AA-108-117 (p6; Rev 2), when equipment protection is required, the equipment being protected, the main power supply feed breaker, and the instrumentation, which if tripped, would render the protected equipment unavailable must be posted. According to TQ-TM-104-623-C001 (p23; Rev 3), Source Range Channel NI-12 is powered from VBB. Consequently, VBB must be protected.

B. **Incorrect.** This is plausible because according to Technical Specification 3.3.1.4 (p3-22; Amendment 263), the reactor shall not be made critical unless two NS Pumps are OPERABLE. However, the plant is in a mode where the Technical Specification does not apply. If the operator incorrectly believed that both NS-P1A and NS-P-1C needed to be protected, then it would be logical to assume that their power supplies needed to be protected as well. According to TQ-TM-104-531-C001 (p26; Rev 6), NS-P-1A ids powered from 480V AC ES Bus 1P, and NS-P-1C is powered from 480V AC ES Bus 1S.

C. **Incorrect.** This is plausible because TQ-TM-104-661-C001 (p107; Rev 4) RM-A-2 must be operable for RCS leakrate determination per Technical Specification 3.1.6.8. The operator may incorrectly believe that since the RB Sump level indicator is OOS, RM-A-2 must be protected. However, According to Technical Specification 3.1.6.8 (p3-12; Amendment 271) RM-A-2 is not required to be operable unless reactor power is > 2%.

D. **Incorrect.** This is plausible because according the Technical Specification 3.1.13 (p3-18g; Amendment 186), the reactor vents must be operable when the reactor is critical.

Technical Reference(s):	OP-AA-108-117 (p4 and 6; Rev 2) Technical Specification 3.5.1 (p3-27 and 3-29; Amendment 189 and 3-30; Amendment 247) TQ-TM-104-623-C001 (p23; Rev 3) TQ-TM-104-661-C001 (p107; Rev 4) Technical Specification 3.1.6.8 (p3-12; Amendment 271) Technical Specification 3.3.1.4 (p3-22; Amendment 263) Technical Specification 3.1.13 (p3-18g; Amendment 186)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 735-GLO-13 (As available)

Question Source:	Bank #	IR-735-GLO-13-Q01	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. This is accomplished by presenting the operator with several components that are out of service, in a given set of plant conditions, and then asking the operator to determine which component must be protected. In answering the question the operator will demonstrate knowledge of the protection process, which not only requires that the in service component be protected but its power supply as well.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information (i.e. power supplies need to be protected, power supplies for SR Instrument, Tech Spec requirements) and then apply them to a given set of plant conditions to answer the question correctly.

What MUST be known:
1. When must an SSC be protected? 2. What are the TS requirements for SR Channel operability under the stated plant conditions? 3. What is the power supply to N-12? 4. How is a SSC protected?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.13	
	Importance Rating	3.4		

Radiation Control: Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: RO Question # 70

Plant conditions:

- 100% power.
- MAP C-1-1, RADIATION LEVEL HI, alarms.
- Letdown Monitor RM-L-1 HIGH CHANNEL is in HIGH ALARM.

Based on this, which ONE (1) of the following identifies the required action?

A. Adjust Seal Injection flow to 26 gpm.

B. Verify that Letdown Isolation Valve MU-V-3 is CLOSED.

C. Evacuate all personnel in the Controlled Area to the HP Checkpoint.

D. Initiate a plant shutdown IAW 1102-4, Power Operation, and 1102-10, Plant Shutdown.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because it is expected that Letdown will be isolated in response, and in accordance with OP-TM-MAP-C0101 (p43-45; Rev 1) Step 4.8.2, the operator is directed to MINIMIZE seal injection while maintaining greater than or equal to 26 gpm using MU-V-32. However, this action is not taken until the alarm has been validated, which has not taken place yet.

B. **Incorrect.** This is plausible because according to OP-TM-MAP-C0101 (p43-45; Rev 1) Step 4.2, the operator is directed to verify that MU-V-2A and MU-V-2B are closed, if the Hi Alarm is Lit or came in and cleared on RM-L-1. So it is expected that Letdown will be

isolated. However, the interlock between RM-L-1 and the Letdown system is associated with MU-V-2A and 2B, not MU-V-3. However, MU-V-3 has some automatic closures. According to TQ-TM-104-211-C001 (p44; Rev 4) Letdown Isolation Valve, MU-V-3 will close on a Loss of Instrument Air, on high letdown temperature 135°F on MU5-TI, on Letdown Line Rupture High Temperature Isolation at ~145°F on MU-TS-1, and on "A" Engineered Safeguards Actuation.

C. **Correct.** According to OP-TM-MAP-C0101 (p43-45; Rev 1) Step 4.1, the operator is directed to announce "High activity has been detected in letdown. All personnel in the controlled area report to H.P. Checkpoint," over the plant page and radio.

D. **Incorrect.** This is plausible because according to OP-TM-MAP-C0101 (p43-45; Rev 1) Step 4.3, the operator is directed to IAAT Pressurizer level is greater than 315", then initiate a plant shutdown IAW 1102-4, Power Operation and 1102-10, Plant Shutdown. The same instruction is provided at Step 4.8.3, however, this action will not be taken because the alarm must be validated first.

Technical Reference(s): OP-TM-MAP-C0101 (p43-45; Rev 1)
TQ-TM-104-211-C001 (p44; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-7 and 10 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the operator must demonstrate knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms; specifically the action to evacuate personnel on high activity detected in the Letdown line.

The question is at the Comprehension/Analysis cognitive level because the operator must select the correct answer among four possible choices that are either included directly, or have an equivalent step (i.e. MU-V-2A/2B), within the pertinent Alarm Response Procedure. By selecting the correct answer, the operator demonstrates an understanding of the strategy of the response actions.

What MUST be known:
1. Evacuation of personnel is required. 2. Isolation of Letdown is required, however not with MU-V-3. 3. Plant shutdown is required conditionally if Pzr level is high, or after the alarm is validated. 4. Seal adjustment is required after the alarm is validated.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.14	
	Importance Rating	3.4		

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 71

Plant conditions:

- 100% power.
- A Steam Generator Tube Leak occurs in the B OTSG.
- RM-A-5, Condenser Vacuum Pump Exhaust, goes into alarm.

Which ONE (1) of the following identifies the LOWEST general area radiation reading in the Turbine Building that will require a plant shutdown if exceeded?

A. 5 mr/hr

B. 10 mr/hr

C. 50 mr/hr

D. 100 mr/hr

Proposed Answer: A

Explanation (Optional):

A. **Correct.** According to OP-TM-MAP-C0101 (p9; Rev 1) Step 4.8, IAAT Turbine Building general area exceeds 5 mr/hr, then INITIATE a plant shutdown IAW 1102-4, "Plant Operation" and 1102-10, "Plant Shutdown".

B. **Incorrect.** This is plausible because according to OP-TM-MAP-C0101 (p9; Rev 1) Step 4.6, IAAT radiation levels are >5 mrem/hr at 10 ft from the Powdex Vessels, then INITIATE OP-TM-423-401, "Backwashing and Precoating a Powdex Vessel - With PBR" for applicable vessels. The operator may incorrectly remember the wrong number.

C. **Incorrect.** This is plausible because it is a reasonable dose rate between two identified

thresholds of OP-TM-MAP-C0101.

D. **Incorrect.** This is plausible because according to OP-TM-MAP-C0101 (p9; Rev 1) Step 4.7, IAAT radiation levels are >100 mr/hr at 10 ft from the Powdex Vessels or Powdex Recovery Vessels, then the operator must shutdown the plant and fix the SGTL. The operator may incorrectly remember the wrong number.

Technical Reference(s): OP-TM-MAP-C0101 (p9; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. The specific knowledge is the threshold for general area radiation readings in the Turbine Building on a SGTL.

The question is at the Memory cognitive level because the operator must recall bits of information to correctly answer the question.

What MUST be known:
1. Lowest General Area Dose Rate in Turbine Building that will require a plant shutdown on a SGTL.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4 2.4.47	
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: RO Question # 72

Plant conditions:

- A LOCA has occurred.
- The crew has entered OP-TM-EOP-009, HPI Cooling, and is verifying ADEQUATE HPI.

The following is observed:

- RCS Pressure is 950 psig.
- Total HPI flow is 410 gpm.
- SCM is 28°F.

Which ONE of the following correctly assesses this situation AND identifies the proper actions?

A. HPI is adequate;
Stop all but one Reactor Coolant Pump, and open the PORV (RC-RV-2).

B. HPI is adequate;
Stop all Reactor Coolant Pumps, and de-energize the Pressurizer Heaters.

C. HPI is NOT adequate;
Stop all but one Reactor Coolant Pump, and open the PORV (RC-RV-2).

D. HPI is NOT adequate;
Stop all Reactor Coolant Pumps, and de-energize the Pressurizer Heaters.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** 1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may interpret the Attachment incorrectly, and determine that adequate HPI exists; and the operator may confuse the strategies of EOP-009.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because these actions would be taken if HPI was adequate. The operator may confuse the strategies of EOP-009.

D. **Correct.** 1st part correct, 2nd part correct. According to OP-TM-EOP-0091 (p3; Rev 1) Step 3.4 is accomplished to verify an adequate makeup supply is available for HPI COOLING. Adequate HPI is verified when HPI flow exceeds flow on Attachment 7.4 in OP-TM-211-901. Consequently, this attachment is provided for the operator to make this determination. Using the intersection of RCS pressure at 950 psig and Total HPI flow of 410 gpm, the operator will determine that HPI flow is NOT adequate. According to OP-TM-EOP-009 (p3 and 5; Rev 6) Step 3.4, if HPI flow is inadequate, the operator is directed to Section 4.0 of EOP-009 where the operator will shutdown all RCPs based on SCM ≥ 25°F, and de-energize all the Pressurizer Heaters.

Technical Reference(s): OP-TM-EOP-009 (p3 and 5; Rev 6)
OP-TM-EOP-0091 (p3; Rev 1)
Attachment 7.4 of OP-TM-211-901 (p22; Rev 6)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: Attachment 7.4 of OP-TM-211-901 (p22; Rev 6)

Learning Objective: EOP009-PCO-2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate the ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. This is accomplished by providing the operator with a set of plant indications that would require the use of an Attachment available in the Control Room to make a determination on ECCS status; and then based on the status, take the appropriate action.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then evaluate plant conditions properly to correctly answer the question.

What MUST be known:
1. Adequate HPI is determined by Attachment 7.4. 2. How is Attachment 7.4 used? 3. What procedure actions are taken within EOP-009 if it is determined that Inadequate HPI flow exists.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4 2.4.1	
	Importance Rating	4.6	

Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: RO Question # 73

Which ONE (1) of the following identifies a condition for which the reactor operator is required to immediately stop the only operating Decay Heat Removal Pump?

A. Cold Shutdown condition with RCS filled, DH SYS VIBS HI (PRF1-4-8) alarm actuates due to high pump vibration.

B. RCS in Refueling Shutdown condition, FUEL TRANS CANAL HI-LO (PLB-4-9) alarm actuates due to low canal level.

C. Cold Shutdown condition with RCS filled, Decay Heat Pump DH-P-1A pump bearing temperature now exceeds Plant Computer HI-1 limits.

D. RCS in Refueling Shutdown condition, unexpectedly incore temperatures have risen 12°F, and RCS water level is below DH pump vortex limit.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.** Operator is not required to immediately trip the pump for this event, in accordance with the alarm response. According to OP-TM-PRF1-0408 (p1; Rev 3), the response actions may ultimately require DH Pump shutdown. However, there are several manual actions that are required prior to taking this action. Distracter is plausible because it is desirable to limit/avoid operation of equipment with high vibration indication.

B. **Incorrect.** The operator is not required to trip the DHR pump for this condition. According to PLB-4-9 (p2; Rev 15), if RCS/DH leakage > 1 gpm, then INITIATE OP-TM-AOP-060, "Leakage While On Decay Heat Removal". According to OP-TM-AOP-060 (p1; Rev 4), there are NO immediate actions to this procedure. Distracter is plausible because Fuel Transfer Canal inventory is low.

C. **Incorrect.** According to TQ-TM-104-212-C001 (p46; Rev 8) the DH Pump A/B Bearing Temperature is monitored by computer points A0734 and A0738; and has two alarm setpoints. The 130°F setpoint (HI-1) indicates an abnormally high temperature but the temperature is within design. The 180°F (HI-2) setpoint indicates the bearing design limit is being exceeded and continued operation will result in bearing failure. Since both alarm setpoints have been exceeded, the condition for taking the DH Pump to PTL has been met.

D. **Correct.** Reference OP-TM-EOP-030, Loss of Decay Heat Removal, entry conditions and immediate actions. According to OP-TM-EOP-030 (p1; Rev 3), the procedure is entered when the Reactor shutdown and both of the following conditions exist: (1) Decay Heat removal is in service, and (2) Incore temperature increases > 10°F due to an unplanned condition. Therefore, based on the stated conditions in the correct answer (D), the procedure would be entered. When the procedure is entered the Immediate Actions must be performed. According to the Immediate Actions if the RCS water level below DH pump vortex limit,, the operating DHR pump must be taken to PTL.

Technical Reference(s):	OP-TM-EOP-030 (p1; Rev 3) TQ-TM-104-212-C001 (p46; Rev 8) PLB-4-9 (p2; Rev 15) OP-TM-AOP-060 (p1; Rev 4) OP-TM-PRF1-0408 (p1; Rev 3)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP030-C001 PCO 2 and 6 (As available)

Question Source:	Bank #	IR-EOP-030-PCO-2-Q01	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: None

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

The KA is matched because the operator must demonstrate knowledge of EOP entry conditions and immediate action steps. This is accomplished by presenting the operator with several sets of plant conditions a requiring that the operator identify the one set of conditions that will permit entry into EOP-030, and placing the DH Pump in PTL.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information (i.e. entry conditions to EOP-030 and IMAs), and then analyze four different sets of conditions, all of which may ultimately require the stopping of the Pump, and choosing the one that requires Immediately stopping the pump.

What MUST be known:
1. What are the entry conditions of EOP-030? 2. What are the IMAs associated with EOP-030? 3. Why does low level in the Fuel Transfer Canal not necessarily require immediate stopping of the DH Pump? 4. Why does high vibration of the operation DH Pump not necessarily require immediate stopping of the DH Pump? 5. Why does a rise in Motor Stator temperature on the operating DH Pump not necessarily require immediate stopping of the DH Pump?

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.36	
	Importance Rating	3.0		

Conduct of Operations: Knowledge of procedures and limitations involved with core alterations.

Proposed Question: RO Question # 74

Plant conditions:

- Refueling Operations in progress.
- DH Loop A in operation.
- RCS temperature at the "A" Decay Heat Removal Pump, DH-P-1A, suction is 130°F and steady.

Event:

- RCS Temperature at the "A" Decay Heat Removal Pump suction has risen 6°F, and continues to rise at 1°F every 15 minutes.

Assuming the RCS temperature continues to rise consistently, identify the ONE selection below that describes the maximum time allowed to continue core alterations.

A. 1 Hour

B. 2 Hours

C. 16 Hours

D. 17 Hours

Proposed Answer: A

Explanation (Optional):

A. **Correct.**

Tech Spec 1.2.6 REFUELING SHUTDOWN

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to

a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

Given an initial "A" Decay Heat Removal Pump, DH-P-1A, suction temperature of 130°F and a rise of 6°F, the difference between 140 and 136 is 4°F. With a rise of 1°F every 15 minutes times 4°F, the "A" Decay Heat Removal Pump, DH-P-1A, suction temperature will reach 140°F in 60 minutes, or 1 Hour.

B. **Incorrect.**

If the examinee recognizes the Tech Spec requirement of 140°F "A" Decay Heat Removal Pump, DH-P-1A, suction temperature, but fails to account for the rise of 6°F in temperature, the difference between 140 and 130 is 10°F. With a rise of 1°F every 15 minutes times 10°F, the "A" Decay Heat Removal Pump, DH-P-1A, suction temperature will reach 140°F in 2.5 Hours. Since this falls between 2 Hours and 3 Hours, and the question is asking maximum time, 2 Hours would be plausible.

C. **Incorrect.**

Incorrect but plausible if the examinee believes that RCS temperature is the highest concern.

Tech Spec 1.2.1 COLD SHUTDOWN

The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is no more than 200°F. Pressure is defined by Specification 3.1.2.

Given an initial RCS temperature of 130°F and a rise of 6°F, the difference between 200 and 136 is 64°F. With a rise of 1°F every 15 minutes times 64°F, the RCS temperature will reach 200°F in 16 Hours.

D. **Incorrect.**

If the examinee recognizes the Tech Spec requirement of 200°F RCS temperature, but fails to account for the rise of 6°F in temperature, the difference between 200 and 130 is 70°F. With a rise of 1°F every 15 minutes times 0°F, the RCS temperature will reach 200°F in 17.5 Hours. Since this falls between 17 Hours and 18 Hours, and the question is asking maximum time, 17 Hours would be plausible.

Technical Reference(s): Tech Spec (p1-1; Amend 157) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	212-GLO-14		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5	
	55.43		

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

The KA is matched because the operator must demonstrate knowledge of limitations involved in core alterations. The knowledge that must be demonstrated is the maximum temperature allowed for performing core alterations.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information and then apply this information to a set of plant conditions to correctly answer the questions.

What MUST be known:
1. What is the maximum temperature allowed per Tech Specs for performing core alterations? 2. The math to figure out time until a temperature limit is reached.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.39	
	Importance Rating	3.9		

Equipment Control: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question: RO Question # 75

Plant conditions:

- Reactor Startup is in progress.
- Reactor Power is 7%.
- BOTH Intermediate Range NI channels are exhibiting erratic behavior and are failing low.
- Troubleshooting and repair will be completed in approximately 12 hours.

In accordance with Technical Specifications which ONE (1) of the following actions, if any, is required?

A. NO action is required. Intermediate Range channels are NOT required above 2% power.

B. Reactor power must be held stable below 10% until at least ONE Intermediate Range channel is returned to service.

C. Within ONE hour, initiate action to place the unit in HOT SHUTDOWN.

D. Within ONE hour, initiate action to verify the operability of at least TWO Power Range detectors, or be in Hot Shutdown within ONE hour.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.** This is plausible because 2% is the value that the reactor enters POWER OPERATION, and the operator may incorrectly believe that once power level has been raised to >2%, the Intermediate Ranges are no longer needed.

B. **Incorrect.** This is plausible according to Technical Specification Table 3.5-1 (p3-30;

Amendment 189), when 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required. The operator may confuse the 10% power concept.

C. **Correct.** In accordance with Technical Specification 3.5.1, at least ONE IR is required below 10% power. With minimum channels not met, action is required IAW Technical Specification 3.0.1. In accordance with Technical Specification 3.5.1 (p3-27, Amendment 189), the reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5-1, Column "A" and "B" are met, except as provided in Table 3.5-1, Column "C". Specification 3.0.1 applies. According to Technical Specification Table 3.5-1 (p3-29; Amendment 247), a minimum of one Intermediate Range Channel must be OPERABLE. Since both instruments are failing the requirement is not met. According to Technical Specification Table 3.5-1 (p3-30; Amendment 189), the operator must restore at least one Intermediate Range within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours. Since it is known that repairs will take 12 hours, the operator must initiate a plant shutdown to Hot Standby within 1 hour.

D. **Incorrect.** This is plausible according to Technical Specification Table 3.5-1 (p3-30; Amendment 189), when 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required. The operator may confuse the 10% power concept.

Technical Reference(s): Technical Specification 3.5.1 (p3-27 and 3-30, Amendment 189) Technical Specification Table 3.5-1 (p3-29; Amendment 247) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO 14 (As available)

Question Source: Bank # WTSI 67924/IS-623-14-Q01

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
NOTE: Question on Cert exam 05-1 (IS level item but entry to TS 3.0.1 and operability requirement for NIS, including modes of operation, considered RO)

The KA is matched because the operator must demonstrate knowledge of less than one hour technical specification action statements for systems (i.e. RPS Instrumentation).

The question is at the Comprehension/Analysis cognitive level because the operator must assemble information (i.e. When and number of IR instruments required), evaluate plant conditions (i.e. repairs will take 12 hours) and determine that conditions for entry into T.S. 3.0.1 apply.

What MUST be known:
1. With the reactor critical at least one Intermediate Range instrument is required to be OPERABLE, unless 2 of 4 Power Range channels indicate > 10%. 2. If no IR channels are OPERABLE, the operator must restore one to service within 1 hour, or the plant must be moved to Hot Standby within the following 6 hours (i.e. Technical Specification 3.0.1 applies).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	062	AA2.02
	Importance Rating		3.6

Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss.

Proposed Question: SRO Question # 76

Plant Conditions:

- Reactor operating at 100% power with ICS in full automatic.
- A drought and heat wave have resulted in rising river water temperature and lowering river water level.
- Operating crews have been in OP-TM-AOP-005, River Water System Failures, for several days.

Event:

- An Auxiliary Operator reports the following from performing his rounds:
  - River water temperature is 91F.
  - ISPH Pump Bay water level is 269ft.

Based on the above conditions and IAW OP-TM-AOP-005, identify (1) the actions the CRS will direct, and (2) the reason for the direction.

A. (1) Trip the reactor, initiate OP-TM-EOP-001, Reactor Trip, and then trip all four Reactor Coolant Pumps.
(2) River Water temperature is too high.

B. (1) Trip the reactor, initiate OP-TM-EOP-001, Reactor Trip, and then trip all four Reactor Coolant Pumps.
(2) ISPH Pump Bay water level is too low.

C. (1) Initiate a plant shutdown IAW 1102-4, Power Operations, to be at Cold Shutdown within thirty six (36) hours.
(2) River Water temperature is too high.

D. (1) Initiate a plant shutdown IAW 1102-4, Power Operations, to be at Cold Shutdown within thirty six (36) hours.
(2) ISPH Pump Bay water level is too low.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct.

Part 2: VERIFY river water temperature < 90ºF, RNO: GO TO Step 4.1. Candidate may choose this answer if they think that the actions in part (1) are due to High River Water Temperature.

B. **Correct.**

Part 1: OP-TM-AOP-005, Step 3.1: IAAT all NR and SR pumps are inoperable or ISPH pump bay water level < 271', then: TRIP the reactor, INITIATE OP-TM-EOP-001 Reactor Trip, and GO TO Section 4.3. (Step 4.3.1 TRIP all four reactor coolant pumps).

Part 2: UFSAR, Section 2.6.3 and TQ-TM-104-168-C001, Section IX.C.1: The lowest expected water level (Elevation 271') would occur from a postulated failure of the center section of the York Haven Hydro Dam. To ensure a flow of water to the intake structure during the lowest expected water level, an intake ditch has been excavated to the deepest channel in the river. It serves the Unit 1 pump house intake structure and has been sized to pass more than maximum required flow to safely shutdown the plant under any emergency condition. The availability capacity is 33% above the maximum emergency capacity allowing for the loss of the York Haven Dam and 188% above the emergency capacity with normal water level. The intake structure was constructed at an elevation to take water from the bottom of the river and to maintain minimum submergence on the nuclear service pumps at all times (minimum 270' above sea level). The low flow intake canal assures that the intake structure has continuous access to river waters at water surface elevations of 270 feet or above even if the main dam at York Haven were removed/damaged.

C. **Incorrect.**

Part 1 is incorrect. OP-TM-AOP-005, Step 3.4:

IAAT ISPH pump bay water level < 274', then INITIATE a plant shutdown IAW 1102-4 "Power Operations" to be at HSD IAW TS 3.0.1 requirements.

Plausible if the candidate does not know the pump bay level required to perform the action. Although the condition is met, the level is too low and has reached another threshold level, requiring a reactor trip IAW Step 3.1. River water level is too low to provide for a safe shutdown and a reactor trip is required instead.

Part 2: VERIFY river water temperature < 90ºF, RNO: GO TO Step 4.1. Candidate may choose this answer if they think that the actions in part (1) are due to High River Water Temperature.

D. **Incorrect.**

Part 1 is incorrect. OP-TM-AOP-005, Step 3.4:

IAAT ISPH pump bay water level < 274', then INITIATE a plant shutdown IAW 1102-4 "Power Operations" to be at HSD IAW TS 3.0.1 requirements.

Plausible if the candidate does not know the pump bay level required to perform the action. Although the condition is met, the level is too low and has reached another threshold level, requiring a reactor trip IAW Step 3.1. River water level is too low to provide for a safe shutdown and a reactor trip is required instead.

However, the level is too low and has reached another threshold level, requiring a reactor trip IAW Step 3.1. River water level is too low to provide for a safe shutdown and a reactor trip is required instead.

Part 2 is correct.

Technical Reference(s): OP-TM-AOP-005 (p1, 11; Rev 9)
UFSAR, p 2.6-7 (Rev 20) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-005-PCO-1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: 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 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	AA2.02
	Importance Rating		3.6

Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm.

Proposed Question: SRO Question # 77

Plant Conditions:

- A reactor coolant leak has occurred and the crew has entered OP-TM-AOP-050, Reactor Coolant Leakage, and OP-TM-EOP-010, GUIDE 9, RCS Inventory Control.
- MU-V-3, RCS LETDOWN RB ISOL VALVE, has been closed.
- Pressurizer Level is being maintained without HPI.

Event:

- The following alarms have actuated:
  - A-1-7, BATT 1A DISCHARGING
  - A-2-7, BATT CHARGER 1A/1C/1E TROUBLE
  - A-3-7, INVERTER 1A/1C/1E TROUBLE
  - L-1-3, VOLTAGE REGULATOR DC LOSS
  - PRF1-1-1, CRD BREAKER TEST TROUBLE
- Auxiliary Operator reports A and C voltages are 104VDC locally and lowering slowly.
- The crew has tripped the reactor.

Based on the above conditions, identify the one selection below that
(1) describes the appropriate action to take, and
(2) the basis for the action.

A. (1) Initiate HPI IAW OP-TM-211-901, per Guide 9.
(2) MU-V-3 fails open on a loss of "A" DC.

B. (1) Initiate HPI IAW OP-TM-211-901, per Guide 9.
(2) A loss of 1M DC Panel occurs on a loss of "A" DC.

C. (1) Close MU-V-2A and MU-V-2B, "A" and "B" L/D COOLER OUT VLV, per OP-TM-AOP-023.
(2) MU-V-3 fails open on a loss of "A" DC.

D. (1) Close MU-V-2A and MU-V-2B, “A” and “B” L/D COOLER OUT VLV, per OP-TM-AOP-023.
(2) A loss of 1M DC Panel occurs on a loss of “A” DC.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.**

Part 1 is incorrect: OP-TM-EOP-010, Guide 9, Step C8: After isolating letdown, the effect on pressurizer level is evaluated. The time spent evaluating should be commensurate with the variance of pressurizer level from the desired trend. If pressurizer level is not being restored (i.e. pressurizer level is not rising) after letdown is isolated and all readily available means of normal makeup have been applied, then HPI is initiated IAW OP-TM-911-901 "Emergency Injection". In this case, Letdown is not isolated since MU-V-3 fails open on the loss of “A” DC, and therefore HPI should not be initiated until Letdown is again isolated and an analysis is performed on Pressurizer level trend from that point on.

Part 2 is correct.

B. **Incorrect.**

Part 1 is incorrect: OP-TM-EOP-010, Guide 9, Step C8: After isolating letdown, the effect on pressurizer level is evaluated. The time spent evaluating should be commensurate with the variance of pressurizer level from the desired trend. If pressurizer level is not being restored (i.e. pressurizer level is not rising) after letdown is isolated and all readily available means of normal makeup have been applied, then HPI is initiated IAW OP-TM-911-901 "Emergency Injection". In this case, Letdown is not isolated since MU-V-3 fails open on the loss of “A” DC, and therefore HPI should not be initiated until Letdown is again isolated and an analysis is performed on Pressurizer level trend from that point on.

Part 2 is incorrect. OP-TM-AOP-023 addresses a Loss of 1M DC in Attachment 4. However, MU-V-18 and MU-V-20 are the Valves powered by 1M DC Bus associated with the Makeup System. Additionally, 1M DC has a transfer switch which will maintain power upon a loss of only one side of DC. Plausible if the candidate incorrectly believes that MU-V-18 closes on a loss of 1M DC and therefore isolates makeup to the pressurizer, requiring HPI.

C. **Correct.**

Part 1: OP-TM-AOP-023, Step 3.15: 3.15 IAAT the “A” or “C” “Battery Voltage” meter lowers to <105 VDC (CB 322’ on “A” Battery Ground Detector), then perform the following: 1. If MU-V-3 is Closed, then ENSURE the following valves are Closed: MU-V-

2A and MU-V-2B.

Part 2: OP-TM-AOP-0231, Step 3.15: If voltage degrades to this level, efforts to restore battery charging capability have been unsuccessful. Battery bank "A" or "C" voltage less than 105 VDC may result in equipment damage, even if the other battery bank voltage is normal. In addition, opening the battery disconnect at 105 VDC prevents discharging the battery to the point that battery damage occurs. This step will de-energize the 1A DC Distribution panel. If MU-V-3 has been closed for some reason, guidance is provided to ensure MU-V-2A and MU-V-2B are closed, since MU-V-3 will fail open on loss of DC.

D. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect. OP-TM-AOP-023 addresses a Loss of 1M DC in Attachment 4. However, MU-V-18 and MU-V-20 are the Valves powered by 1M DC Bus associated with the Makeup System. Additionally, 1M DC has a transfer switch which will maintain power upon a loss of only one side of DC. Plausible if the examinee believes that MU-V-2A and MU-V-2B fail open and require to be closed.

Technical Reference(s): OP-TM-AOP-023 (p9; Rev 3)
OP-TM-AOP-0231 (p8; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-023-PCO-1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: 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 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	065	AA2.08
	Importance Rating		3.3

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment.

Proposed Question: SRO Question # 78

Plant conditions:

- Reactor power is operating at 100% with ICS in full automatic.
- Backup Instrument Air Compressor, IA-P-2A, is tagged out of service for maintenance.
- Main Vacuum Pump, VA-P-1A, is operating.

Event:

- 0100 Local fire (now extinguished) renders SA-P-1A and SA-P-1B inoperable.
- 0200 PLB-1-6, IA-P-4/IA-Q-2 TROUBLE, actuates due to IA-Q-2 high d/p.
- 0210 IA-P-1A trips on motor overcurrent.
- 0220 IA-P-1B trips on motor overcurrent.
- 0230 The following valid alarms actuate:
  - PLB-1-7, INSTRUMENT AIR PRESS LOW TURBINE AREA.
  - PLB-1-8, STATION SERVICE AIR PRESSURE LOW.
  - PLB-2-7, INSTRUMENT AIR PRESS LOW AUX BLDG AREA.

Currently (0240)

- All Instrument and Service Air System pressure indications are 65 psig and lowering.
- Main Condenser vacuum is 27 inches Hg, reducing at 0.2 inches Hg per minute.

Based on these conditions, identify (1) the components below which are inoperable at this time (0240), and (2) the appropriate Technical Specification action statement.

A. (1) All Emergency Feedwater Pumps due to EFW Control Valves to the "A" and "B" OTSG's, EF-V-30A/B/C/D, failing closed.
(2) Maintain 100% reactor power until EFW is restored.

B. (1) All Emergency Feedwater Pumps due to EFW Control Valves to the "A" and "B" OTSG's, EF-V-30A/B/C/D, failing closed.
(2) Place the Unit in Hot Shutdown in 24 Hours.

C. (1) Both Emergency Diesel Generators due to the Diesel Generator Room "A" and "B" fan dampers, AH-E-29A and AH-E-29B, failing closed.
(2) Place the Unit in Hot Shutdown in 12 Hours.

D. (1) Both Emergency Diesel Generators due to the Diesel Generator Room "A" and "B" fan dampers, AH-E-29A and AH-E-29B, failing closed.
(2) Place the Unit in Hot Shutdown in 24 Hours.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect:

IAW OP-TM-AOP-028, Attachment 6.1, Effects of loss of IA on Safety Related or Critical Power Production Air operated Equipment, page 1 of 5:

EF-V-30A and EF-V-30C have a backup air supplied by 2 Hour Air System – Train "A".
EF-V-30B and EF-V-30D have a backup air supplied by 2 Hour Air System – Train "B".

IAW TQ-TM-104-850-C001:

- Two Hour Backup Instrument Air System provides dry compressed air to critical components of the Main Steam, Emergency Feedwater, and Reactor River Systems for two (2) hours, should Instrument Air or Backup Instrument Air NOT be available.

- Technical Specification definition 1.3 Operable
  - "Shall be OPERABLE…when…necessary instrumentation, controls, …or…other auxiliary equipment…(is) also capable of performing (its) related support function(s)."

- Two Hour Backup Instrument Air required to satisfy Technical Specification 3.4 Decay Heat Removal (DHR) Capability
  - Provides motive force for EF-V-30s, MS-V-4s and MS-V-6

- Two Hour Backup Instrument Air can supply 2 hours of air if at 1500 psig
  - Minimum 1700 psig is specified in the procedure to provide margin
    - Assures sufficient allowance of leakage to be detected and corrected so that the minimum inventory requirement is not jeopardized (1500 psig).

Although all four valves will fail closed on a loss of air, this will not occur for another 2 hours. Plausible if the candidate believes that the valves will fail closed immediately.

Part 2 is incorrect but plausible since this would be the correct answer if EF-V-30A-D were inoperable.

T.S. 3.4.1, Note 1: Specification 3.0.1 and all other actions requiring shutdown or changes in REACTOR OPERATING CONDITIONS are suspended until at least two EFW Pumps and one EFW flowpath to each OTSG are restored to OPERABLE status.

B. **Incorrect.**
Part 1 is incorrect:
IAW OP-TM-AOP-028, Attachment 6.1, Effects of loss of IA on Safety Related or Critical Power Production Air operated Equipment, page 1 of 5:
EF-V-30A and EF-V-30C have a backup air supplied by 2 Hour Air System – Train "A".
EF-V-30B and EF-V-30D have a backup air supplied by 2 Hour Air System – Train "B".

Part 2 is incorrect but plausible if the candidate believes that T.S. 3.4.1.1.a.(3) applies:

T.S. 3.4.1.1.a.(3):
With one main steam supply path to the turbine-driven EFW Pump and one motor-driven EFW Pump inoperable, restore the steam supply or the motor-driven EFW Pump to OPERABLE status within **24 hours** or be in HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 12 hours.

C **Correct:**
Part 1:
AH-E-29A is the "A" Diesel Generator Room Fan.
AH-E-29B is the "B" Diesel Generator Room Fan.

IAW OP-TM-AOP-028, Attachment 6.1, Effects of loss of IA on Safety Related or Critical Power Production Air operated Equipment, page 5 of 5, the associated dampers for AH-E-29A and AH-E-29B fail closed. This renders AH-E-29A and AH-E-29B inoperable.

Part 2:
OP-TM-AOP-028, Attachment 6.1, Effects of loss of IA on Safety Related or Critical Power Production Air operated Equipment, page 5 of 5 (An associated note at the end of the attachment states):

*Without an operable AH-E-29A/B, EG-Y-1A/B is inoperable. Enter TS 3.7.2.C action statement.

TS 3.7.2.C
Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days* provided that the redundant diesel generator is:

1. verified to be operable immediately;
2. within 24 hours, either:
   a. determine the redundant diesel generator is not inoperable due to a common mode failure; or
   b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24

hours thereafter.

D **Incorrect.**
Part 1 is correct:

Part 2 is incorrect but plausible if the candidate believes that the Technical Specification states **24 Hours**.

TS 3.7.2.C
Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days* provided that the redundant diesel generator is:

1. verified to be operable immediately;
2. within **24 hours**, either:
   a. determine the redundant diesel generator is not inoperable due to a common mode failure; or
   b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional **24 hour** period the plant shall be placed in COLD SHUTDOWN within an additional **24 hours** thereafter.

Technical Reference(s): Tech Spec (p3-43; Amend 258) OP-TM-AOP-028 (p31; Rev 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-028-PCO-14 (As available)

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

Question History: 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	2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	077	2.4.4
	Importance Rating		4.7

Generator Voltage and Electric Grid Disturbances: Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (Generator Voltage and Electric Grid Disturbances).

Proposed Question: SRO Question # 79

Plant conditions:

- Reactor power is 50% with ICS in full automatic.
- Power escalation is in progress following a refueling outage.

Event:

- The following alarms have actuated simultaneously and then clear:
  - L-1-5 MAIN XFMR OVER-VOLTAGE
  - H-1-4 NEUTRON X-LIMIT TO FW
  - H-2-2 LARGE MW ERROR IN TRACK
  - H-1-3 MN TURB HDR PRESS HI/LO
  - B-2-2 4KV ES BUS UV/OV
  - B-2-4 480V ES BUS UV/OV
- The following alarms have actuated simultaneously and remain lit:
  - H-3-3 MN TURB ON MANUAL
  - H-2-1 ICS IN TRACK
  - H-1-6 OTSG A LO LVL LIMIT
  - H-1-7 OTSG B LO LVL LIMIT
- MWe reduces to 48, and stabilizes.
- Reactor Power reduces to 14%, and stabilizes.
- GB1-12 and GB1-02 indicate closed.
- All breakers on panel SS-1 indicate open.

Based on these conditions, identify the ONE selection below that describes:

(1) The HIGHEST PRIORITY procedure to respond to these conditions.
(2) Required action.

A. (1) OP-TM-EOP-001, Reactor Trip.

(2) Trip the Reactor and Turbine.

B. (1) OP-TM-AOP-022, Load Rejection.
(2) Start and load EG-Y-1A on the 1D 4160V Bus.

C. (1) OP-TM-424-901, Emergency Feedwater.
(2) Verify EFW actuation.

D. (1) OP-TM-301-471, Manual Control of the Main Turbine.
(2) Return the turbine to auto.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** Plausible if the symptoms are misinterpreted as a loss of offsite power; however there are still MWe and the Generator Breakers are closed.

B. **Correct.** The Substation Breakers are open and the Generator Breakers are closed supplying the plant with power. AOP-022 requires starting and loading EG-Y-1A to satisfy TS 3.7.

C. **Incorrect.** Plausible if the symptoms are misinterpreted. The OTSG Low Level limit alarms would be expected at this power level and if the running Feedwater Pump had tripped the reactor and turbine would have tripped.

D. **Incorrect.** Plausible since the turbine is in manual and will be returned to auto; however it is not the highest priority procedure, AOP-022, Load Rejection would be the highest priority.

Technical Reference(s): OP-TM-AOP-022, Load Rejection. (p3; Rev 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-022-PCO-2 (As available)

Question Source: Bank # ILT 10-01 Audit Exam #82

Modified Bank # (Note changes or attach parent)

New

Question History:		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	5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	054	2.2.37
	Importance Rating		4.6

Loss of Main Feedwater: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: SRO Question # 80

Initial Conditions:

- Reactor tripped on a loss of both main feedwater pumps, FW-P-1A and FW-P-1B.
- Due to equipment failures at time of the trip, no Emergency Feedwater Pumps are operating to supply EFW to the OTSGs.
- Condensate Booster Pumps, CO-P-2A/B/C, are not operating.
- Decay Heat Removal system is NOT operating.
- Subcooling Margin is 35°F and lowering.
- CRS transitions to OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer, due to Incore temperatures rising.

Currently:

- CO-P-2A has just been started.
- Preparations to restore Main Feedwater have just been initiated.
- Primary to Secondary Heat Transfer has not been established.
- RCS is approaching 25°F Subcooling Margin, currently 26°F and lowering rapidly.
- OTSG Pressures are 1000 psig and steady.

Based on these conditions identify the ONE selection below that describes:

(1) Method of core cooling to be established.
(2) Applicable procedure.

A. (1) HPI cooling.
(2) OP-TM-EOP-009, HPI Cooling.

B. (1) Rapid cooldown to LPI injection.
(2) OP-TM-EOP-002, Loss of 25 Degrees F Subcooling Margin.

C. (1) Condensate Booster Pump feed.
(2) OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer.

D. (1) Normal Main Feedwater feed.
(2) OP-TM-401-103, Shifting FW-P-1A from Standby Mode to Operating Mode.

Proposed Answer: A

Explanation (Optional):

A. **CORRECT** answer. Since the crew is already in OP-TM-EOP-004, Step 3.6 states: IAAT RCS is approaching 25 °F SCM, then GO TO EOP-009.

IAW OP-TM-EOP-004 Basis Document:

If SCM is not adequate, then exit EOP-004 and initiate HPI COOLING IAW EOP-009. This is one of two routes to EOP-009. This one provides the primary path when feedwater is available, and SCM is lost during attempts to restore primary to secondary heat transfer. If feedwater is available, but primary to secondary heat transfer has not been established, then efforts to restore heat transfer may continue until plant subcooling margin < 25F.

B. **INCORRECT.** Plausible if the candidate believes that a symptom exists for entry into OP-TM-EOP-002, Loss of 25 °F Subcooling Margin. EOP-002 will direct back to EOP-004. Going to Section 4.0 of EOP-002, Rapid RCS Cooldown, is only if both HPI is inadequate and SCM is greater than 25°F.

IAW OP-TM-EOP-009 Basis Document:

The step intent is to initiate all actions required for a loss of adequate subcooling margin (RCPs are tripped immediately; HPI, LPI and EFW systems are actuated) and not transition to EOP-002. Since the most likely cause of a loss of subcooling margin in EOP-009 is opening the PORV, **transfer to EOP 002, "Loss of 25 °F Subcooled Margin" is not desired**. Before opening the PORV, adequate HPI flow is verified, and as long as the action from Rule 1 is performed, no other action from EOP 002 would be required.

C. **INCORRECT.**
OP-TM-EOP-004, Step 3.10:
If all of the following conditions are met:
- A Condensate Booster Pump is On
- At least one RCP is On
- An OTSG is intact

then PERFORM Attachment 1, "OTSG Feed Using a Condensate Booster Pump".

Although a Condensate Booster Pump is running and the conditions are met, this method requires forced RCS flow and will not be effective as soon as Subcooling Margin is less than 25°F. RCP's must be secured within one minute of a Loss of Subcooling Margin and the conditions will no longer be met to perform Attachment 1.

OP-TM-EOP-004, Step 3.6 states:

IAAT RCS is approaching 25 °F SCM, then GO TO EOP-009.

D. **INCORRECT.**
The method of core cooling identified is not correct for stem conditions The candidate may believe that Main Feedwater can be started once vacuum is restored, but procedurally they must go to EOP-009. The procedure, although correct for starting Main Feedwater, is not the correct procedure to use for the given conditions.

Technical Reference(s): OP-TM-EOP-004 (p 3; Rev 7) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: 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 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	038	2.4.30
	Importance Rating		4.1

Steam Generator Tube Rupture: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: SRO Question # 81

Which ONE of the following events would require four (4) hour report to the NRC Operations Center, in accordance with LS-AA-1020, "Reportability Reference Manual"?

A. Activation of ERDS, following declaration of an ALERT.

B. Initiating a plant shutdown for an OTSG tube leak (0.5 gpm leak rate).

C. Cold shutdown loss of offsite power (LOOP), both diesels load onto busses.

D. Completion of a cooldown to replace a leaking Code Safety Valve (0.5 gpm leak rate).

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.** Plausible activation of ERDS requires notification, time however is 1 hour.

LS-AA-1020 Page 2:

Emergency Response Data System (ERDS) activation. As soon as possible, but not later than 1 hour after declaring an Alert, Site Area Emergency, or General Emergency.

B. **Correct.** An OTSG tube leak >150 gpd is a Condition of License in Technical Specifications requiring Shutdown, per F-aa of LS-AA-1020 10CFR50.72(b)(2)(i) would apply.

LS-AA-1020 Page 4:

Initiation of any nuclear plant shutdown required by the Technical Specifications. ENS within 4 hours. Written report required by 10CFR50.73 if shutdown is completed.

C. **Incorrect.** Plausible would be an E-plan entry unusual event, would require 1 hour notification.

LS-AA-1020 Page 1:

The declaration of any of the Emergency Classes specified in the Emergency Plan. Immediately by ENS after notification of State and local agencies, but within 1 hour of declaration of Emergency Class. Written report required by 10CFR72.75. Notify within 2 hours of declaring an Alert or higher emergency classification.

D. **Incorrect.** Plausible however cooldown does not require notification unless T.S. initiated.

LS-AA-1020 Page 4:

Initiation of any nuclear plant shutdown required by the Technical Specifications. ENS within 4 hours. Written report required by 10CFR50.73 if shutdown is completed.

Technical Reference(s): LS-AA-1020 (p4; Rev 17) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: LS-AA-1020

Learning Objective: AOP-005-PCO-1 (As available)

Question Source: Bank # IR-XXX-GLO-X-Q82

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	K/A #	A06	AA2.1	
	Importance Rating			4.2

Ability to determine and interpret the following as they apply to the (Shutdown Outside Control Room): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 82

Event:

- A Fire in the Relay Room has occurred.
- The crew has initiated OP-TM-EOP-020, Cooldown From Outside of Control Room. IMA's are complete and Control Room Evacuation has just commenced.
- Intermediate Closed Cooling Water pump, IC-P-1A, has tripped.
- Due to an open circuit at the fire location, Intermediate Closed Cooling Water pump, IC-P-1B, failed to start.
- MU-V-32, RCP Seal Injection Control Valve, inlet flow is inadequate.
- 1E 4KV bus was deenergized but is now reenergized IAW OP-TM-EOP-020, Attachment 9, Starting EG-Y-1B and Loading 1E 4160V Bus.

Based on the above conditions, (1) how will the CRS address Intermediate Closed Cooling Water, and (2) what procedure addresses the action?

A. (1) After ES/UV lockouts are reset, IC-P-1B is required to be started to prevent CRD stator damage.
(2) OP-TM-AOP-032, Loss of Intermediate Closed Cooling Water.

B. (1) After ES/UV lockouts are reset, IC-P-1B is required to be started to prevent CRD stator damage.
(2) OP-TM-EOP-020, Cooldown From Outside of Control Room.

C. (1) After establishing control at the RSD panels, IC-P-1B is required to remain shutdown to prevent RCP thermal barrier cooler damage.
(2) OP-TM-AOP-032, Loss of Intermediate Closed Cooling Water.

D. (1) After establishing control at the RSD panels, IC-P-1B is required to remain shutdown to prevent RCP thermal barrier cooler damage.
(2) OP-TM-EOP-020, Cooldown From Outside of Control Room.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.**

Part (1) The reactor is tripped in the event of loss of cooling to the CRD stators in order to prevent stator damage. Loss of CRD stator cooling would not prevent CRD insertion on RPS actuation. Maintaining reactor shutdown is not affected by loss of IC component cooling.

Part (2) is not correct as OP-TM-AOP-032 does not address actions from outside of the Control Room.

B. **Incorrect.**

Part (1) The reactor is tripped in the event of loss of cooling to the CRD stators in order to prevent stator damage. Loss of CRD stator cooling would not prevent CRD insertion on RPS actuation. Maintaining reactor shutdown is not affected by loss of IC component cooling.

Part (2) is correct. OP-TM-AOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel.

C. **Incorrect.**

Part (1) is correct.

Part (2) is not correct as OP-TM-AOP-032 does not address actions from outside of the Control Room.

D. **Correct.**

Part (1) is correct. EOP-0201 Step 3.2

If EG-Y-1B was required to be started IAW Attachment 9 (i.e., the bus was de-energized), then it is assumed that RCP seals may be overheated. Seal return would be isolated, IC-P-1B stopped to prevent thermal barrier cooler damage, and OP-TM-226-901 is initiated.

Part (2) is correct. OP-TM-AOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel.

Technical Reference(s): OP-TM-EOP-020 (p 81; Rev 13) (Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	EOP-020-PCO-1		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		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	5	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	037	AA2.09
	Importance Rating		3.4

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: System status, using independent readings from redundant Condensate air ejector exhaust monitor.

Proposed Question: SRO Question # 83

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": 6.4 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 910 psig and steady; Level is 40% and rising slowly.
- OTSG "B" Pressure is 885 psig and steady; Level is 50% and rising slowly.
- RCS Pressure is 925 psig and lowering.
- BWST level is 26 feet and lowering at 1 foot every 30 minutes.
- Offsite dose assessor reports in that projected offsite integrated dose is 0.5 R whole body and 1.5 R thyroid.

Based on the above plant conditions, what action is required to be directed by the CRS IAW OP-TM-EOP-005?

A. Isolate OTSG "A", ONLY.

B. Isolate OTSG "B", ONLY.

C. Isolate OTSG "A" and OTSG "B".

D. Continue steaming OTSG "A" and OTSG "B".

Proposed Answer: B

Explanation (Optional):

There are select isolation criteria within OP-TM-EOP-005, OTSG TUBE LEAKAGE:

- 3.34 IAAT OTSG level > 85% Operate Range, then perform the following:
    - 1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the OTSG.
    - 2. If both OTSGs are being isolated, then GO TO EOP-009

- 3.35 IAAT both OTSGs are available, and projected or actual offsite integrated dose approaches 0.5 R whole body or 1.5R thyroid, then perform the following:
    - 1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the most affected OTSG.

- 3.59 IAAT BWST level < 22 ft, then perform the following:
    - 1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the most affected OTSG.

- 3.60 IAAT BWST level < 15 ft, then perform the following:
    - 1. If both OTSGs are affected, then INITIATE Attachment 1A or 1B to isolate the second OTSG.

- 3.64 When RCS temperature is being controlled by DHR, or incore temperature < 200°F, then PERFORM Attachment 1A or 1B to isolate the affected OTSG.

A. **Incorrect.**

"A" OTSG does not meet isolation criteria IAW OP-TM-EOP-005, but the choice is plausible if the candidate believes that it is the most affected OTSG based on radiation monitor readings. Although the number appears larger, it is E3 compared to E4 for the "B" OTSG. See Explanation above.

B. **Correct.**

OP-TM-EOP-005, Step 3.35

IAAT both OTSGs are available, and projected or actual offsite integrated dose approaches 0.5 R whole body or 1.5R thyroid, then perform the following: 1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to **isolate the most affected OTSG**.

"B" OTSG is the most affected OTSG based on radiation monitor readings.

C. **Incorrect.**

Will be chosen if the candidate believes that upon a report of projected offsite integrated dose being 0.5 R whole body, that any/all OTSG's with a primary to secondary tube leak must be isolated. Plausible since there are conditions within OP-TM-EOP-005 to isolate both OTSG's. None of those conditions are met however. See Explanation above.

D. **Incorrect.**

Plausible if the candidate believes that 1.5 R is the threshold for whole body instead of 0.5 R for whole body and 1.5 R for thyroid. See Explanation above.

Technical Reference(s): OP-TM-EOP-005 (p17; Rev 8)
T.S. (p 3-1a - 3.2, Amend 266) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-005-PCO-1 (As available)

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

Question History: 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 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	A03	2.1.23
	Importance Rating		4.4

Loss of NNI-Y - Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO Question # 84

Plant conditions:

- The plant is at 70% power.
- Feedwater Pump Status:
  - o FW-P-1A is running.
  - o FW-P-1B is OOS.
- Due to an ICS problem that is being investigated, the following ICS stations are in HAND:
  - o A and B Feedwater Loop Masters
  - o FW-V-17B, B Main Feed Water Control Valve
  - o FW-V-16B, B FW Start Up Control Valve
  - o MS-V-3A/B/C/4B

Event:

- MAP H-1-8, ICS/NNI POWER LOST, actuates.
- At Panel PCL:
  - ICS-HAND ICS/NNI Power indicator lamp is NOT lit.
  - SUBFEEDS AUTO/HAND Power indicator lamp is NOT lit.
- The following alarms actuate with no change in alarm related parameters:
  - C-2-7, DH PUMP SUCTION TEMP HI.
  - G-3-8, RC PRESS NARROW RNG HI/LO.

Based on these conditions, identify (1) the appropriate action to be performed by the crew, and (2) the appropriate Tech Spec timeclock.

A. (1) Enter OP-TM-AOP-026, Loss of ATB or ICS HAND Power to perform Reactor Trip IMAs, trip FW-P-1A, initiate Emergency Feedwater, and then initiate OP-TM-EOP-001, Reactor Trip.
(2) Thirty (30) Days.

B. (1) Enter OP-TM-AOP-026, Loss of ATB or ICS HAND Power to perform Reactor Trip IMAs, trip FW-P-1A, initiate Emergency Feedwater, and then initiate OP-TM-EOP-001, Reactor Trip.
(2) Seventy-Two (72) Hours.

C. (1) Enter OP-TM-AOP-025, Loss of ICS HAND and AUTO Power, to perform Reactor Trip IMAs, trip FW-P-1A, initiate Emergency Feedwater, and then initiate OP-TM-EOP-001, Reactor Trip.
(2) Thirty (30) Days.

D. (1) Enter OP-TM-AOP-025, Loss of ICS HAND and AUTO Power, to perform Reactor Trip IMAs, trip FW-P-1A, initiate Emergency Feedwater, and then initiate OP-TM-EOP-001, Reactor Trip.
(2) Seventy-Two (72) Hours.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 Correct.

Part 2 is incorrect but plausible if the candidate believes that a loss of ATB has occurred, which is also an entry for AOP-026. However, based on entry conditions, a loss of ATB has not occurred:

AOP-026, Section 1.0 Entry Conditions, Note:

Loss of ATB is evident by loss of power to OWS, PPC monitors and MAP G-2-6 illuminated.

AOP-026, ATTACHMENT 1

ATB: The following controls cannot be transferred to RSD Panel:

- IC-P-1B,
- MS-V-8A, MS-V-8B,
- NR-V-1B, NR-V-15B,
- RC-V-2, RC-V-3
- Communications

Tech Spec 3.5.7 States:

The minimum number of functions identified in Table 3.5-4 shall be OPERABLE. With the number of functions less than the minimum required, restore the required function to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within an additional 12 hours (RC-V-2, RC-V-3, MS-V-8A, MS-V-8B are listed in Table 3.5-4)

B. **Correct.**

Part 1: Both lamps (loss of Hand and loss of Subfeeds) are NOT LIT, therefore a Loss of Hand Power has occurred, Loss of AUTO has not been identified as being NOT LIT, so a Loss of ICS AUTO has not occurred.

AOP-026, Step 2.1:

IAAT reactor power < 75% and either FW-V-17A or FW-V-17B is in HAND, then perform the following:

1. PERFORM EOP-001, "Reactor Trip" Immediate Actions.
2. TRIP both Main Feedwater pumps:
   ___ FW-P-1A
   ___ FW-P-1B
3. INITIATE OP-TM-424-901, "Emergency Feedwater".
4. INITIATE EOP-001, "Reactor Trip".

Part 2: AOP-026, Step 3.4:

If any Turbine Bypass valves are in HAND, then perform the following:

1. Press the ICS ADV "B/U LOADER" pushbutton.
2. Close associated isolation valve:
   -MS-V-8A
   -MS-V-8B
3. Request SM to evaluate TS 3.4.1.1.b timeclock entry.

Tech Spec 3.4.1.1.b:

Four of six Turbine Bypass Valves (TBVs) OPERABLE. With more than two TBVs inoperable, restore operability of at least four TBVs within 72 hours.

Since MS-V-3A/B/C were in HAND control, they now cannot be controlled from the Control Station and are therefore INOPERABLE, and therefore the Tech Spec is applicable.

C. **Incorrect.**

Part 1 is incorrect but plausible if the candidate confuses the indications with a Loss of AUTO and HAND power and chooses to enter AOP-025.

Part 2 is incorrect but plausible if the candidate believes that a loss of ATB has occurred, which is also an entry for AOP-026. However, based on entry conditions, a loss of ATB has not occurred:

AOP-026, Section 1.0 Entry Conditions, Note:

Loss of ATB is evident by loss of power to OWS, PPC monitors and MAP G-2-6 illuminated.

AOP-026, ATTACHMENT 1

ATB: The following controls cannot be transferred to RSD Panel:

- IC-P-1B,
- MS-V-8A, MS-V-8B,
- NR-V-1B, NR-V-15B,

- RC-V-2, RC-V-3
- Communications

Tech Spec 3.5.7 States:

The minimum number of functions identified in Table 3.5-4 shall be OPERABLE. With the number of functions less than the minimum required, restore the required function to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within an additional 12 hours (RC-V-2, RC-V-3, MS-V-8A, MS-V-8B are listed in Table 3.5-4)

D. **Incorrect.**

Part 1 is incorrect but plausible if the candidate confuses the indications with a Loss of AUTO and HAND power and choose to enter AOP-025.

Part 2 is correct.

Technical Reference(s): OP-TM-AOP-026 (p1,3; Rev 3)
Tech Spec (p3-26; Amend 242) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-026-PCO-1 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Original Question:

Plant conditions:

- The plant is at 68% power
- Feedwater Pump 1A is running and FW-P-1B is OOS for maintenance
- The Feedwater Loop Masters are in HAND
- FW-V-17B and FW-V-16B are in HAND due to an ICS problem that is being investigated

Event:

- ICS HAND Power is lost

With the above conditions the crew will ____________.

A. GO TO OP-TM-EOP-001, Reactor Trip, and trip the reactor only.

B. GO TO OP-TM-EOP-001, Reactor Trip, and trip the reactor, PERFORM the VSSVs and transition to OP-TM-AOP-026, Loss of ATB or ICS HAND Power.

C. INITIATE OP-TM-AOP-026, Loss of ATB or ICS HAND Power and stabilize the plant at the current power level with the ICS in HAND.

D. INITIATE OP-TM-AOP-026, Loss of ATB or ICS HAND Power, Perform OP-TM-EOP-001, Reactor Trip IMAs, trip FW-P-1A, INITIATE Emergency Feedwater and INITIATE OP-TM-EOP-001, Reactor Trip.

Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	005	2.2.38
	Importance Rating		4.5

Inoperable/Stuck Control Rod - Knowledge of conditions and limitations in the facility license.

Proposed Question: SRO Question # 85

Plant Conditions:

- •Reactor is operating at 50% power, BOL, with ICS in automatic.
- •All 4 RCP's are operating.
- •Due to a stuck rod, Indicated Rod Index is 100% Withdrawn.

Based on the above conditions, identify the ONE selection below that describes:
(1) Actions required for the limit violation, and
(2) The maximum time allowed IAW Tech Specs to achieve acceptable Control Rod Index.

A. (1) Lower RCS Boron Concentration to withdraw rods.
(2) 2 Hours.

B. (1) Lower RCS Boron Concentration to withdraw rods.
(2) 24 Hours.

C. (1) Raise RCS Boron Concentration to withdraw rods.
(2) 2 Hours.

D. (1) Raise RCS Boron Concentration to withdraw rods.
(2) 24 Hours.

Proposed Answer: D

Explanation (Optional):

The examinee will need to recognize that COLR Figure 1 (page 1 of 2) is the applicable figure for Beginning of Life. At 60% power and 100% Withdrawn Rod Index, the examinee recognizes operation within the restricted Range.

A. **Incorrect**

Part 1 is incorrect because boron would need to be raised to withdraw rods if the rod insertion limit is not met. Plausible if the candidate does not understand the relationship between boron and rod index limit.

Part 2 is incorrect. Plausible if the examinee uses COLR Figure 1 (page 2 of 2) and therefore inaccurately determines that operation is within the Not Allowed Region.

Tech Spec 3.5.2.5 Control Rod Positions

> b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.
>
> > 2. If regulating rods are .inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within **2 hours** or reduce power to ≤ power allowed by rod insertion limits.

B. **Incorrect**

Part 1 is incorrect because boron would need to be raised to withdraw rods if the rod insertion limit is not met. Plausible if the candidate does not understand the relationship between boron and rod index limit.

Part 2 is correct.

C. **Incorrect**

Part 1 is correct.

Part 2 is incorrect. Plausible if the examinee uses COLR Figure 1 (page 2 of 2) and therefore inaccurately determines that operation is within the Not Allowed Region.

Tech Spec 3.5.2.5 Control Rod Positions

> b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.
>
> > 2. If regulating rods are .inserted in the unacceptable operating region, initiate boration within 15 minutes to restore SDM to ≥1% ΔK/K, and restore regulating rods to within restricted region within **2 hours** or reduce power to ≤ power allowed by rod insertion limits.

D. **Correct**

Part 1 is correct. Examinee must realize that boron would need to be raised to withdraw rods if the rod insertion limit is not met.

1102-4, Power operations

3.5 Control Rod Withdrawal to Maintain Desired Rod Limits
3.5.1 Prerequisites

- Group 7 Control Rod position is more than 5% below the desired band

3.5.2 Procedure

1. **Raise RCS boron concentration as required to withdraw control rods to within the limits:**
   1. DETERMINE amount of borated water to add IAW ReMA reactivity data sheets.
   2. ADD the boric acid IAW the following procedures:
      - OP-TM-211-455 "Feed from RCBT"
      - OP-TM-211-456 "Feed from the BAMT"
      - OP-TM-211-457 "Feed from an RBAT"
      - OP-TM-211-458 "Feed from one RCBT and BAMT"
      - OP-TM-211-459 "Feed & Bleed"

Part 2 is correct.
Tech Spec 3.5.2.5 Control Rod Positions

b. Position limits are specified for regulating control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified in the CORE OPERATING LIMITS REPORT.

1. If regulating rods are inserted in the restricted operating region, corrective measures shall be taken immediately to achieve an acceptable control rod position. **Acceptable control rod positions shall be attained within 24 hours,** and FQ(Z) and F(ΔH/N) shall be verified within limits once every 2 hours, or power shall be reduced to ≤ power allowed by insertion limits.

Technical Reference(s): T.S 3.5.2.5 (p3-35; Amend 273)
1102-4 (p23; Rev 121)
COLR (p 13; Rev 8)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: COLR Figure 1

Learning Objective: 622-GLO-14 (As available)

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

Question History: Last NRC Exam: 2005

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43 2	

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	006		A2.10
	Importance Rating			3.9

Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low boron concentration in SIS

Proposed Question: SRO Question # 86

Initial Plant Conditions:

- Reactor operating at 100% power.
- BWST Boron Concentration is 2550ppm.
- BWST Water Temperature is 50°F.

Event:

- Chemistry reports the latest BWST sample results as follows:
  - BWST Boron Concentration is 2475ppm.
  - BWST Water Temperature is 45°F.

What is currently (1) the immediate concern, and (2) the action required?

A. (1) Boron crystallization.
(2) BWST water temperature must be restored to limits within **8 hours** IAW OP-TM-212-457, BWST Heater Operations.

B. (1) Boron crystallization.
(2) BWST water temperature must be restored to limits within **1 hour** IAW OP-TM-212-457, BWST Heater Operations.

C. (1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident.
(2) BWST boron concentration must be restored to limits within **8 hours** IAW OP-TM-212-527, Transfer of C RCBT to the BWST.

D. (1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident.
(2) BWST boron concentration must be restored to limits within **1 hour** IAW OP-TM-212-527, Transfer of C RCBT to the BWST.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.**

Part 1 is incorrect. Boron crystallization is a concern with low BWST water temperature. T.S. minimum temperature is 40°F. Initial conditions stated a temperature greater than the T.S. limit.

Part 2 is correct.

B. **Incorrect.**

Part 1 is incorrect. Boron crystallization is a concern with low BWST water temperature. T.S. minimum temperature is 40°F. Initial conditions stated a temperature greater than the T.S. limit.

Part 2 is incorrect. 72 Hours is the Timeclock associated with the Makeup Tank. Plausible because it is also within T.S. 3.3.1.1.

C. **Correct.**

Part 1: T.S. 3.3 Basis:

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). **The limits on BWST minimum volume and boron concentration ensure that** 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) **the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).**

Part 2: T.S. 3.3.1.1 Injection Systems

a. The borated water storage tank **(BWST)** shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, **restore the BWST to OPERABLE within 8 hrs**. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

D. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect. 72 Hours is the Timeclock associated with the Makeup Tank. Plausible because it is also within T.S. 3.3.1.1.

Technical Reference(s):	T. S. 3.3.1.1 (p 3-21,3-23; Amend 263)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	212-GLO-14	(As available)

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

Question History: 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	2

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	010	A2.01	
	Importance Rating			3.6

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures.

Proposed Question: SRO Question # 87

Plant conditions:

- Reactor power is 100%, with ICS in full automatic.
- Pressurizer Pressure Control is in automatic.

Event:

- An electrical transient causes the following to occur:
  - o 1P 480V ES Bus Unit 3A, Pressurizer Htr Grp 8 Backup Pwr Supply, is faulted.
  - o Loss of ICS Auto power.
  - o Selected Narrow Range RCS pressure transmitter fails low.
  - o SASS fails to operate.

If no manual actions are taken, then 60 seconds after the event occurs, (1) what will occur, and (2) what Technical Specification action, if any, is applicable?

A. (1) Saturation Pressure in the pressurizer will be lower due to a pressurizer low level interlock.
(2) Restore equipment to operable within 7 days or be in Hot Standby IAW 1102-4 within 6 hours.

B. (1) Saturation Pressure in the pressurizer will be lower due to a pressurizer low level interlock.
(2) No T.S. action as Group 9 heaters can be transferred to 1S 480V ES Bus IAW OP-TM-220-901, Emergency Power Supply For Pressurizer Heaters.

C. (1) Makeup Tank level will remain constant due to MU-V-17, NORMAL MU TO RCS CONTROL VALVE, controller transferring to HAND control.
(2) Restore equipment to operable within 7 days or be in Hot Standby IAW 1102-4 within 6 hours.

D. (1) Makeup Tank level will remain constant due to MU-V-17, NORMAL MU TO RCS CONTROL VALVE, controller transferring to HAND control.
(2) No T.S. action exists as MU-V-17 has transferred to ICS HAND control IAW OP-TM-AOP-027, Loss of ATA or ICS Auto Power.

Proposed Answer: A

Explanation (Optional):

A. **Correct**

Part 1:
AOP-027, Attachment 1, Effects of Loss of ATA and ICS/NNI AUTO Power:
- Pressurizer heaters fail OFF (pressurizer low level interlock actuated in failed state). Without heaters to raise pressure in the pressurizer, Saturation pressure will lower.

Part 2: Tech Spec 3.1.3.4.2(a) states:
With the pressurizer inoperable due to one (1) inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 7 days or be in at least Hot Standby within the next 6 hours and in Hot Shutdown within the following 12 hours.

Since 1P 480V bus is deenergized, Group 8 heaters cannot be powered and therefore the Tech Spec is applicable.

B. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect. OP-TM-220-901 cannot be performed under the given conditions as the reactor is still critical:

3.3. Prerequisites
- VERIFY the Reactor is shutdown.

Additionally, Tech Spec 3.1.3.4.2.a is in effect regardless of Group 9 being energized since 1P 480V bus is deenergized.

C. **Incorrect.**

Part 1 is incorrect because:
1. MU-V-17 fails to last demanded position,
2. Letdown is isolated and therefore MU-T level will decrease.

Plausible for lack of knowledge of letdown component interactions.

Part 2 is correct.

D. **Incorrect.**

Part 1 is incorrect because:
1. MU-V-17 fails to last demanded position,
2. Letdown is isolated and therefore MU-T level will decrease.

Plausible for lack of knowledge of letdown component interactions.

Part 2 is incorrect. Plausible if the student believe that Tech Spec 3.1.3.4.1 is the only applicable one and that MU-V-17 going to HAND will prevent us from lowering below 80 inches.

Tech Spec 3.1.3.4.1:

The reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.

However, Tech Spec 3.1.3.4.2.a is in effect regardless of Group 9 being energized since 1P 480V bus is deenergized.

Technical Reference(s): OP-TM-AOP-027 (p23;Rev 3)
T.S. 3.1.3.4.2.a (p3-6a; Amend 78) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-027-PCO-2 (As available)

Question Source: Bank #

Modified Bank # ILT 10-01 Audit exam, # 24 (Note changes or attach parent)

New

Question History: 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 2

Comments:

Original Question:

Plant conditions:

- Reactor power is 100%, with ICS in full automatic.
- Pressurizer Pressure Control is in automatic.

Event: An electrical transient causes the following to occur:

- Loss of ICS Auto power.
- Selected Narrow Range RCS pressure transmitter fails low.
- SASS fails to operate.

If no manual actions are taken, then 60 seconds after the event occurs:

- Makeup Tank level will be higher due to MU-V-17 traveling closed.

- Makeup Tank level will remain constant due to MU-V-17 controller transferring to HAND control.

- Saturation Pressure in the pressurizer will be lower due to a pressurizer low level interlock.

- Saturation Pressure in the pressurizer will be higher due to pressurizer heater banks 1 through 5 energizing.

Proposed Answer: C.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	012	2.2.12
	Importance Rating		4.1

Knowledge of surveillance procedures.

Proposed Question: SRO Question # 88

Plant Conditions:

- The monthly surveillance test on the "D" RPS channel is in progress IAW 1303-4.1D, RPS Channel D Test.

Event:

- The surveillance has identified that the channel will NOT trip on high temperature.
- Replacement parts are not in stock. They have been ordered, but will not be delivered for several days.

Based on the above conditions, which ONE of the following choices is the MINIMUM actions required IAW 1303-4.1D and Tech Specs?

A. Notify Operations Shift Management, and then perform the entire monthly surveillance on the remaining three RPS channels.

B. Notify Operations Shift Management, and then perform only the affected portion of the monthly surveillance on the remaining three RPS channels.

C. Repair the "D" RPS channel high temperature function, and then perform the entire monthly surveillance on the remaining three RPS channels.

D. Repair the "D" RPS channel high temperature function, and then perform only the affected portion of the monthly surveillance on the remaining three RPS channels.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

The first part is correct in that 1303-4.1D states to notify Operations Shift management, but Tech Spec 4.1 requires that only the associated module be tested in the remaining three RPS cabinets, not the complete monthly tests.

B. **Correct.**

1303-4.1D, Section 3.1 Precautions:
- Off-normal conditions:
  - If at any time the RTM fails to trip when expected, then IMMEDIATELY notify Operations Shift Management.
  - Upon detection of a failure that prevents a trip action in a channel, the remaining RPS channels' instrumentation associated with the protective parameter will be tested (T.S. Paragraph 4.1).

T.S. 4.1 Basis:
Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

C. **Incorrect.**

Neither Tech Spec 4.1 nor 1303-4.1D state to repair the module prior to taking actions required by Tech Specs. Plausible if candidate believes that the procedure directs repairing of the module based on RPS logic. Additionally, Tech Spec 4.1 requires that only the associated module be tested in the remaining three RPS cabinets, not the complete monthly tests.

D. **Incorrect.**

Neither Tech Spec 4.1 nor 1303-4.1D state to repair the module prior to taking actions required by Tech Specs. Plausible if candidate believes that the procedure directs repairing of the module based on RPS logic. The second part is correct in that only the associated module needs to be tested in the remaining three RPS cabinets.

Technical Reference(s): T.S. 4.1 (p 4-2a; Amend 274)
1303-4.1D (p2; Rev 22) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	641-GLO-14		(As available)
Question Source:	Bank #	IR-641-GLO-14-Q03	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		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	2	

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	026	2.4.21	
	Importance Rating			4.6

Containment Spray: 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: SRO Question # 89

Initial Conditions:

- 100% power.
- No maintenance or surveillances in progress.

Event:

- Engineering reports to the Control Room that the latest revised calculations indicate there are 32,000 lbs of trisodium phosphate dodecahydrate (TSP) in the Reactor Building emergency sump.

Based on the above conditions: (1) what is the impact of the report, and (2) which Critical Safety Function is affected by the report?

This amount of TSP affects the....

A. (1) Reactor Building Spray System by increasing the formation of precipitates that may migrate to the emergency sump.
(2) CSF 5, Containment Integrity.

B. (1) Reactor Building Spray System by increasing the formation of precipitates that may migrate to the emergency sump.
(2) CSF 8, Auxiliary Emergency Systems.

C. (1) Makeup System by increasing the potential for chloride induced stress corrosion cracking of austenitic stainless steel.
(2) CSF 5, Containment Integrity.

D. (1) Makeup System by increasing the potential for chloride induced stress corrosion cracking of austenitic stainless steel.
(2) CSF 8, Auxiliary Emergency Systems.

Proposed Answer: A

Explanation (Optional):

A. **Correct:**

Part (1) is correct because Per T.S.:

3.3.1.3 Reactor Building Spray System and Reactor Building Emergency Cooling System: The following components must be OPERABLE:

a. Two reactor building spray pumps and their associated spray nozzles headers and two reactor building emergency cooling fans and associated cooling units (one in each train). Specification 3.0.1 applies.

b. The Reactor Building emergency sump pH control system shall be maintained with **≥ 18,815 lbs and ≤ 28,840** lbs of trisodium phosphate dodecahydrate (TSP). Specification 3.3.2.1 applies.

3.3 Bases (p3-24)

The Reactor Building emergency sump pH control system ensures a sump pH between 7.3 and 8.0 during the recirculation phase of a postulated LOCA.
- A minimum pH level of 7.3 is required to reduce the potential for chloride induced stress corrosion cracking of austenitic stainless steel and assure the retention of elemental iodine in the recirculation fluid.
- A maximum pH value of 8.0 **minimizes the formation of precipitates that may migrate to the emergency sump** and minimizes post-LOCA hydrogen generation. Trisodium phosphate dodecahydrate is used because of the high humidity that may be present in the Reactor Building during normal operation. This form is less likely to absorb large amounts of water from the atmosphere.

Part (2) is correct:

CSF 5, Containment Integrity: Provide means to prevent or minimize fission product release to the environment. (1) Maintain containment pressure below design and (2) Provide capability to isolate the containment when required.

Per TQ-TM-104-214-C001:

The TSP chemical functions to mitigate the consequences of an accident by controlling the RB sump pH. In the event of a LOCA and containment flooding, the pH of the sump water would be maintained between 7.3 and 8.0. Control of post LOCA containment sump pH is important for the following reasons:

- Too low a pH could result in radioactive iodine evolving from the sump water into containment atmosphere thus increasing dose rates inside containment and increasing the potential for control room and offsite doses.
- Too low a pH could contribute to stress corrosion cracking which could lead to failures in safety related equipment or components.
- Too high a pH could result in excessive hydrogen generation in containment, resulting in a potentially explosive atmosphere.

Excessive TSP would drive pH High. Too high a pH could result in a breakdown of insulating material in containment, which could contribute to

containment sump blockage.

B. **Incorrect.**

Part (1) is correct.

Part (2) is plausible if the candidate believes that a high amount of Trisodium Phosphate affects Auxiliary Emergency Systems, or that the Reactor Building Spray System is an Auxiliary Emergency System.

CSF-8: - Provide equipment cooling (closed cooling & ventilation), and other support requirements to accomplish the other Critical Safety Functions.
- Provide Instrument Air for operation of EFW, ADV's, RCP Support Systems, and some containment isolation valves.

C. **Incorrect.**

Part (1) is incorrect but plausible if candidate confuses the bases for the low end of the band for the high end of the band.

Part (2) is correct.

D. **Incorrect.**

Part (1) is incorrect but plausible if candidate confuses the bases for the low end of the band for the high end of the band.

Part (2) is incorrect but plausible if the candidate believes that a low amount of Trisodium Phosphate affects Auxiliary Emergency Systems.

CSF-8: - Provide equipment cooling (closed cooling & ventilation), and other support requirements to accomplish the other Critical Safety Functions.
- Provide Instrument Air for operation of EFW, ADV's, RCP Support Systems, and some containment isolation valves.

TSP affects none of these listed items.

Technical Reference(s): Tech Spec (p 3-22, 3-23, 3-24; Amend 263, ECR TM 09-00160) OS-24 (p32; Rev 19) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GLO-214-14 (As available)

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

Question History: 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	2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013	A2.04
	Importance Rating		4.2

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument bus.

Proposed Question: SRO Question # 90

Refer to the diagram provided on the following page.
Plant Conditions:

- 100% Power.
- ESAS Relay 63Y1/RC1A (1600 PSIG) is deenergized due to an electrical fault.

Event:

- A loss of Vital Bus B occurs.

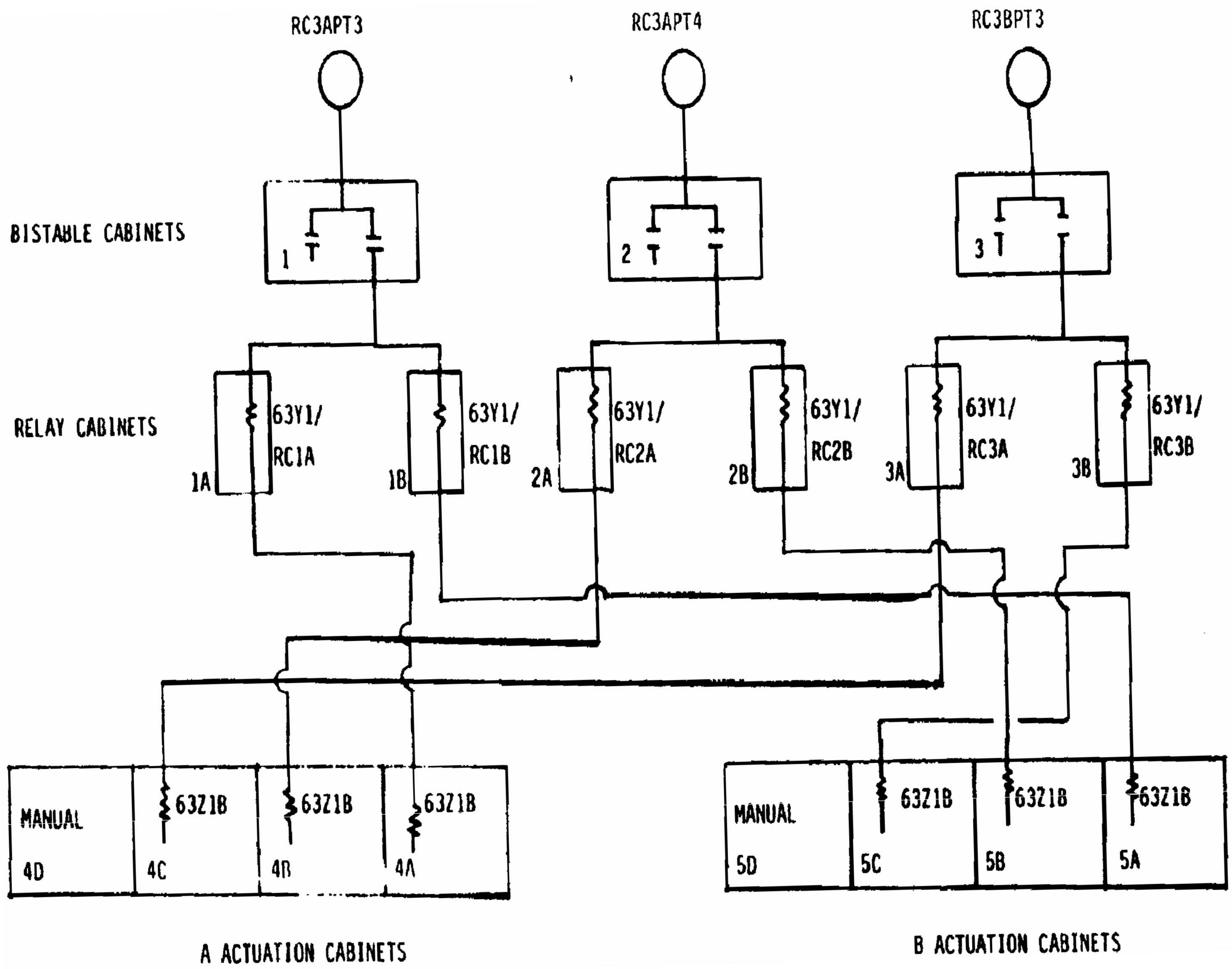
Which of the following describes:
(1) The response of the Emergency Safeguards Actuation System (ESAS), and
(2) Which is the required action that the CRS will direct?

A. (1) ESAS Channel II bistable cabinet de-energizes causing an ACTUATION of "A" Train of ESAS, ONLY.
(2) Place MU-V-17 in Hand IAW OP-TM-AOP-016, Loss of VBB.

B. (1) ESAS Channel II bistable cabinet de-energizes causing an ACTUATION of "A" Train of ESAS, ONLY.
(2) Shutdown Makeup Pump, MU-P-1A, IAW OP-TM-AOP-046, Inadvertent ESAS Actuation.

C. (1) ESAS Channel II Relay cabinets lose power causing an ACTUATION of the "A" and "B" Trains of ESAS.
(2) Place MU-V-17 in Hand IAW OP-TM-AOP-016, Loss of VBB.

D. (1) ESAS Channel II Relay cabinets lose power causing an ACTUATION of the "A" and "B" Trains of ESAS.
(2) Shutdown Makeup Pumps, MU-P-1A and MU-P-1C, IAW OP-TM-AOP-046, Inadvertent ESAS Actuation.



RC3APT3
RC3APT4
RC3BPT3
BISTABLE CABINETS
1
2
3
RELAY CABINETS
63Y1/ RC1A
63Y1/ RC1B
63Y1/ RC2A
63Y1/ RC2B
63Y1/ RC3A
63Y1/ RC3B
1A
1B
2A
2B
3A
3B
MANUAL
4D
63Z1B
4C
63Z1B
4B
63Z1B
4A
MANUAL
5D
63Z1B
5C
63Z1B
5B
63Z1B
5A
A ACTUATION CABINETS
B ACTUATION CABINETS

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect. OP-TM-AOP-016, Loss of VBB, Step 3.2:

If RC2-TE2 is selected for pressurizer level compensation, then perform the following:

A. PLACE MU-V-17 in HAND (CC).
B. CONTROL pressurizer level IAW OP-TM-211-472, Manual Pressurizer Level Control".

Since RC2-TE1 is the preferred instrument and nothing to the contrary was given in the stem, then the step in OP-TM-AOP-016 is actually not applicable.

Plausible if the student believes that Pressure Level Compensation was lost upon the loss of VBB.

B. **Correct**.

Part 1 is correct because:

- Relay 63Y1/RC1A sends an actuation signal to one of the three "A" Actuation Cabinets of ESAS, therefore one channel of ESAS is in on the A side only.
- A loss of Vital Bus B causes RC3APT4 to deenergize, which in turn deenergizes #2 Bistable Cabinet, which in turn deenergizes 63Y1/RC2A and 63Y1/RC2B. This sends an actuation signal to one of the three Actuation Cabinets of both "A" and "B" ESAS, therefore two channels are now in on the "A" side and 1 channel is now in on the "B" side. A side has actuated, B does not meet the 2/3 logic and therefore is not actuated.

Part 2 is correct because:

A) OP-TM-AOP-046 Entry Conditions have been met:
All of the following:
– Makeup and Purification System in the ES Standby Mode.
– Any of the following:
  – ESAS 1600# RCS press actuation.
  – ESAS 500# RCS press actuation.
  – ESAS 4# RB press actuation.
– No ESAS actuation setpoints have been exceeded.
– No fires in the following zones:
  – AB-FZ-4: AB 281' Shield Wall Area
  – CB-FA-3C: ESAS Room
  – CB-FA-3D: Relay Room
  – CB-FA-4B: Control Room

B) OP-TM-AOP-046 Immediate Manual Action states to defeat invalid ESAS signals. However, the signals will not be able to be defeated. The RNO for this step is to Go To Section 4.0.

C) The first step of Section 4.0 is to Shutdown "A" train Makeup Pump not required for seal injection. MU-P-1B is still operating, so MU-P-1A will need to be shut down.

C. **Incorrect.**

Part 1 is incorrect because only one channel of "B" ESAS has actauted, and therefore a 2/3 logic has not been met. Plausible if the student believes that two cabinets have actuated on each side of ESAS.

Part 2 is incorrect. OP-TM-AOP-016, Loss of VBB, Step 3.2:

If RC2-TE2 is selected for pressurizer level compensation, then perform the following:

A. PLACE MU-V-17 in HAND (CC).
B. CONTROL pressurizer level IAW OP-TM-211-472, Manual Pressurizer Level Control".

Since RC2-TE1 is the preferred instrument and nothing to the contrary was given in the stem, then the step in OP-TM-AOP-016 is actually not applicable.

Plausible if the student believes that Pressure Level Compensation was lost upon the loss of VBB.

D. **Incorrect.**

Part 1 is incorrect because only one channel of "B" ESAS has actauted, and therefore a 2/3 logic has not been met. Plausible if the student believes that two cabinets have actuated on each side of ESAS.

Part 2 is incorrect as only one channel of "B" ESAS has actauted, and therefore a 2/3 logic has not been met. Since "B" ESAS has not actuated, MU-P-1C has not started.

Plausible if the student believes that two cabinets have actuated on each side of ESAS.

MU-P-1C included in the choice for plausibility with part (1).

Technical Reference(s):	OP-TM-AOP-046 (p1,17; Rev 0) OP-TM-AOP-016 (p17; Rev 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: Attached drawing only

Learning Objective:	642-GLO-5		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		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	5	
Comments:			

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	K/A #	016	A2.01	
	Importance Rating			3.1

Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: SRO Question # 91

Plant conditions:

- Reactor is operating at 100%, with ICS in full automatic.
- RPS Cabinet B is de-energized for maintenance.

Sequence of events:

- Due to a slow failure, a false signal output reduction is occurring from RPS Channel A RCS Pressure transmitter.
- RPS Channel A is placed in Manual Bypass, RPS Channel A RCS pressure indication continues to LOWER slowly.
- ACTUAL RCS pressure is now 2260 psig, RISING slowly due to automatic Pressurizer heater operation.

Based on these conditions, identify the ONE selection below that describes:

(1) Impact of the malfunction on operation of the Spray Valve **and** PORV.
(2) The procedural actions required to mitigate the consequences of the failure.

A. (1) AUTOMATIC operation of the Spray Valve ONLY is Failed.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

B. (1) AUTOMATIC operation of the Spray Valve ONLY is Failed.
(2) Turn off Pressurizer heaters and fully open the Spray Valve IAW OP-TM-MAP-G0308, RC PRESS NARROW RNG HI/LO.

C. (1) AUTOMATIC operation of the Spray Valve and the PORV is Failed.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

D. (1) AUTOMATIC operation of the Spray Valve and the PORV is Failed.
(2) Turn off Pressurizer heaters and fully open the Spray Valve IAW OP-TM-MAP-G0308, RC PRESS NARROW RNG HI/LO.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.**

(1) Continued reduction in false pressure signal will prevent automatic open operation for the Spray Valve and the PORV. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

(2) OP-TM-AOP-043 is not the correct procedure. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

B. **Incorrect.**

(1) Continued reduction in false pressure signal will prevent automatic open operation for the Spray Valve and the PORV. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

(2) is correct

C. **Incorrect.**

(1) is correct.

(2) OP-TM-AOP-043 is not the correct procedure. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

D. **Correct.**

(1) Continued reduction in false pressure signal will prevent automatic open operation for the Spray Valve and the PORV.

TQ-TM-104-624-C001

Narrow Range RC Pressure, 1700–2500 psig: Rosemount Capacitance Type Detectors.

2 detectors per loop:

RC3A-PT1/2 for the "A" loop (A0586); and
RC3B-PT1/2 for the "B" loop (A0587).

- RC3A-PT1 feeds the "A" RPS cabinet and RC3B-PT1 feeds the "B" RPS cabinet.

- Hi/Lo pressure alarm G-3-8 comes from a relay in RC3-PR for the "A" Hotleg and the bar graph meter for the "B" Hotleg (setpoint 2255/2055).
- Both PT's are SASS monitored and are capable of providing an input to:
    PZR Heaters, both modulating and bistable controlled.
    Spray Valve control.
    PORV High Pressure setpoint control.

With RPS Cabinet B de-energized for maintenance, SASS will not transfer to the B instrument.

(2) OP-TM-MAP-G0308 is the correct procedure to be implemented from the choices given.

4.0 MANUAL ACTIONS REQUIRED
4.1 If RCS pressure is HI, then PERFORM the following:
1. ENSURE pressurizer heaters are Off.
2. Fully OPEN RC-V-1 PZR Spray Control Valve.
3. RETURN RC-V-1 PZR Spray Control Valve to automatic.

Technical Reference(s): OP-TM-MAP-G0308 (p1; Rev 3) TQ-TM-104-624-C001 (p29; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-14 (As available)

Question Source: Bank # IS-624-GLO-14-Q01
Modified Bank # (Note changes or attach parent)
New

Question History: 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	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	068	2.2.40
	Importance Rating		4.7

Liquid Radwaste System (LRS) - Ability to apply Technical Specifications for a system.

Proposed Question: SRO Question # 92

Plant conditions:

- 100% power.
- All controls in normal alignment.
- An approved radioactive liquid release is in progress.

Event:

- Annunciator C-1-1 "RADIATION LEVEL HI" is in alarm.
- Radiation Monitor RM-L-6, RAD WASTE DISCHARGE, has failed (pegged high, off-scale).
- WDL-V-257 is stuck open and cannot be closed.

Based on these conditions, identify the ONE selection below that describes:

(1) Which ONE of the following is the correct action?
(2) What is the timeclock associated with RM-L-6?

A. (1) Terminate the release and notify Radiation Protection.
(2) RM-L-6 must be returned to OPERABLE status within 14 days.

B. (1) Terminate the release and notify Radiation Protection.
(2) RM-L-6 must be returned to OPERABLE status within 30 days.

C. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) RM-L-6 must be returned to OPERABLE status within 14 days.

D. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) RM-L-6 must be returned to OPERABLE status within 30 days.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct because OP-TM-MAP-C0101 states:

RM-L-6, Rad. Waste Discharge

4.1 If RM-L-6 count rate is not lowering, then TERMINATE the liquid release IAW liquid release procedure in use:
- – OP-TM-232-551, "Liquid Release of 'A' WECST with WDL-P-14A"
- – OP-TM-232-552, "Liquid Release of 'A' WECST with WDL-P-14B"
- – OP-TM-232-554, "Liquid Release of 'B' WECST with WDL-P-14B"
- – OP-TM-232-555, "Liquid Release of 'B' WECST with WDL-P-14A"

Part 2 is incorrect because the ODCM states 14 days for an inoperable gas channel:

Action 30.2
If the inoperable gas channel(s) is not restored to service within 14 days, a special report shall be submitted to the Regional Administrator of the NRC Region I Office and a copy to the Director, Office of Inspection and Enforcement within 30 days of declaring the channel(s) inoperable. The report shall describe (a) the cause of the monitor inoperability, (b) action being taken to restore the instrument to service, and (c) action to be taken to prevent recurrence.

Plausible if the candidate confuses the gas channel requirement with the liquid channel requirement, both in the ODCM.

B. **Correct.**

Part 1 is correct because OP-TM-MAP-C0101 states:

RM-L-6, Rad. Waste Discharge

4.1 If RM-L-6 count rate is not lowering, then TERMINATE the liquid release IAW liquid release procedure in use:
- – OP-TM-232-551, "Liquid Release of 'A' WECST with WDL-P-14A"
- – OP-TM-232-552, "Liquid Release of 'A' WECST with WDL-P-14B"
- – OP-TM-232-554, "Liquid Release of 'B' WECST with WDL-P-14B"
- – OP-TM-232-555, "Liquid Release of 'B' WECST with WDL-P-14A"

Part 2 is correct.

ODCM Step 2.1.1.b:
With less than the minimum number of radioactive liquid effluent monitoring

instrumentation channels OPERABLE, take the ACTION shown in Table 2.1-1. Exert best efforts to return the instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Effluent Release Report why the inoperability was not corrected in a timely manner.
* For WDL-FT-84, and RM-L-6, operability is not required when discharges are positively controlled through the closure of WDL-V-257.

C. **Incorrect.**

Part 1 is incorrect.

RM-L-7, Plant Discharge:

4.5 IAAT high alarm is Lit, then perform the following:

- REQUEST Chemistry to collect a sample from SR-V-57 (Rad. Mon. Pit).

Plausible if action is confused as one for RM-L-6.

Part 2 is incorrect because the ODCM states 14 days for an inoperable gas channel:

Action 30.2
If the inoperable gas channel(s) is not restored to service within 14 days, a special report shall be submitted to the Regional Administrator of the NRC Region I Office and a copy to the Director, Office of Inspection and Enforcement within 30 days of declaring the channel(s) inoperable. The report shall describe (a) the cause of the monitor inoperability, (b) action being taken to restore the instrument to service, and (c) action to be taken to prevent recurrence.

Plausible if the candidate confuses the gas channel requirement with the liquid channel requirement, both in the ODCM.

D. **Incorrect.**

Part 1 is incorrect.

RM-L-7, Plant Discharge:

4.5 IAAT high alarm is Lit, then perform the following:

- REQUEST Chemistry to collect a sample from SR-V-57 (Rad. Mon. Pit).

Plausible if action is confused as one for RM-L-6.

Part 2 is correct.

ODCM Step 2.1.2.b:
With less than the minimum number of radioactive gaseous process or effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table

2.1-2. Exert best efforts to return the instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

Technical Reference(s): ODCM (p20,22; Rev 2)
OP-TM-MAP-C0101 (p50; Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO-14 (As available)

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

Question History: 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 2

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	K/A #	029	2.1.20	
	Importance Rating			4.6

Containment Purge System: Ability to interpret and execute procedure steps.

Proposed Question: SRO Question # 93

Plant Conditions:

- The Reactor is Shutdown.
- A Reactor Building Purge was being performed in accordance with OP-TM-823-406, RB Purge – Containment Closed.

Event:

- The Purge was temporarily shutdown while investigating fan vibration.
- The cause was located and corrected.

Which ONE of the following describes the MINIMUM requirement for reinitiating the purge using the existing Gaseous Waste Release permit?

Purge has been shutdown for less than ____(1)____ hours, and ____(2)____ approval has been obtained.

A. (1) 4
(2) Operations Director

B. (1) 4
(2) Radiological Protection

C. (1) 12
(2) Operations Director

D. (1) 12
(2) Radiological Protection

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part (1) is correct.

Part (2) is incorrect. RC will determine whether the existing permit will be used, and they will issue the approval. Plausible if the examinee confuses the requirement per Step 4.12 in 6610-ADM-4250.12:

> If the purge will be performed while the reactor is critical, the Shift Management will obtain the concurrence of the Operations Director prior to commencing the purge

Initial conditions stated that the reactor was shutdown.

B. **Correct Answer**.

Part (1): OP-TM-823-406 States:

> 4.6 If continuing a purge that has been temporarily shutdown, then VERIFY shutdown has been for **< 4 hours** and Radiological Controls has approved continuation under the current Radiological Release Permit.

Part (2): 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents - Reactor Building Purges, states:

> 4.1.C: A Reactor Building purge may be stopped then restarted within 4 hours using the same release permit. If the purge is secured for more than 4 hours, or if conditions change that may warrant resampling as determined by Radiological Protection, a new release permit (with new air samples) will be required.

C. **Incorrect.**

Part (1) is incorrect. 12 Hours is plausible if the candidate confuses the time required to restart a purge with the amount of time the release permit is valid after a sample. Also, the Note prior to Step 4.29 in OP-TM-823-406:

> NOTE: The Waste Gas Release Permit becomes invalid if purge is shut down for more than 4 hours. If for any reason restart does not occur within 4 hours of shutdown, this procedure will be closed out and a new release permit must be obtained. Purge can only be restarted per this procedure if plant conditions have not changed (i.e., RB doors and/or equipment hatch will stay closed and no change to purge valves limits will be made).

Part (2) is incorrect. RC will determine whether the existing permit will be used, and they will issue the approval. Plausible if the examinee confuses the requirement per Step 4.12 in 6610-ADM-4250.12:

> If the purge will be performed while the reactor is critical, the Shift Management will obtain the concurrence of the Operations Director prior to commencing the purge

Initial conditions stated that the reactor was shutdown.

D. **Incorrect.**

Part (1) is incorrect. 12 Hours is plausible if the candidate confuses the time required to restart a purge with the amount of time the release permit is valid after a sample. Also, the Note prior to Step 4.29 in OP-TM-823-406:

> NOTE: The Waste Gas Release Permit becomes invalid if purge is shut down for more than 4 hours. If for any reason restart does not occur within 4 hours of shutdown, this procedure will be closed out and a new release permit must be obtained. Purge can only be restarted per this procedure if plant conditions have not changed (i.e., RB doors and/or equipment hatch will stay closed and no change to purge valves limits will be made).

Part (2) is correct.

Technical Reference(s): OP-TM-823-406 (p 4,8; Rev 9)
6610-ADM-4250.12 (p 3; Rev 15) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 231-GLO-10 (As available)

Question Source: Bank # IR-XXX-GLO-X-Q97
Modified Bank # (Note changes or attach parent)
New

Question History: 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 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.42	
	Importance Rating		3.4

Knowledge of new and spent fuel movement procedures.

Proposed Question: SRO Question # 94

Plant Conditions:

- Unit is in a refueling outage.
- Spent Fuel Shuffles are in progress in the Fuel Handling Building.
- An assembly move is in progress and has currently traveled greater than 50% of the required distance.

Event:

- RM-A-4, Fuel Handling Bldg. Exhaust, is in High Alarm.
- Radiation Protection has been directed to verify the alarm.

(1) When must the Fuel Handling SRO declare Emergency Suspension of Fuel Handling Activities, and (2) what actions must the Fuel Handling SRO direct prior to evacuating the Fuel Handling Building IAW 1505-1, Fuel and Control Component Shuffles?

A. (1) Upon verification of alarm by Radiation Protection.
(2) Return fuel assembly to original location, and maintain the fuel assembly engaged.

B. (1) Upon verification of alarm by Radiation Protection.
(2) Complete the assembly move, disengage the fuel assembly, and raise the grapple to the "Full Up" position.

C. (1) Upon the announcement made regarding RM-A-4.
(2) Return fuel assembly to original location, and maintain the fuel assembly engaged.

D. (1) Upon the announcement made regarding RM-A-4.
(2) Complete the assembly move, disengage the fuel assembly, and raise the grapple to the "Full Up" position.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect**.

Part 1 is incorrect. Although OP-TM-MAP-C0101, Radiation Level High, states to notify Radiation Protection to verify the alarm in Step 4.2, 4.5.5.A states to suspend fuel handling IAW 1505-1, “Fuel and Control Component Shuffles”. There is no requirement to wait for verification from Radiation Protection. Plausible if the candidate believes that suspension can wait based on the Shift Manager questioning the validity of the alarm.

Part 2 is incorrect but plausible if the candidate believes that the minimum action to take is completing the assembly move. This could be chosen based upon the urgency of a Radiation Monitoring alarm and the need to evacuate the Fuel Handling Building. Additionally, the candidate could believe that IAW 1505-1:

> Step 5.3.7: When inserting into core, the fuel assembly shall **NOT** be disengaged from grapple until count rate stability has been verified.

But the conditions state that **Spent Fuel** Shuffles are in progress in the **Fuel Handling Building**.

B. **Incorrect.**

Part 1 is incorrect. Although OP-TM-MAP-C0101, Radiation Level High, states to notify Radiation Protection to verify the alarm in Step 4.2, 4.5.5.A states to suspend fuel handling IAW 1505-1, “Fuel and Control Component Shuffles”. There is no requirement to wait for verification from Radiation Protection. Plausible if the candidate believes that suspension can wait based on the Shift Manager questioning the validity of the alarm.

Part 2 is correct.

C. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect but plausible if the candidate believes that the minimum action to take is completing the assembly move. This could be chosen based upon the urgency of a Radiation Monitoring alarm and the need to evacuate the Fuel Handling Building. Additionally, the candidate could believe that IAW 1505-1:

> Step 5.3.7: When inserting into core, the fuel assembly shall **NOT** be disengaged from grapple until count rate stability has been verified.

But the conditions state that **Spent Fuel** Shuffles are in progress in the **Fuel Handling Building**.

D. **Correct.**

Part 1:
OP-TM-MAP-C0101, Radiation Level High:

RM-A-4 Fuel Handling Bldg. Exhaust:

4.5 IAAT high alarm is Lit, then perform the following: ___

1. ENSURE AH-E-10 is Shutdown.
2. ENSURE the following are Closed:
   - AH-D-120
   - AH-D-121
   - AH-D-122
3. ANNOUNCE the high alarm and that all personnel evacuate the Fuel Handling Building over plant page and radio.
4. Notify Radiation Protection to perform the following:
   A. INITIATE surveys and OBTAIN air samples of affected area.
   B. RESTRICT access to affected area.
5. If fuel handling is in progress, then perform the following:
   A. **NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel and Control Component Shuffles".**

Part 2:

1505-1, "Fuel and Control Component Shuffles

5.3.12 If emergency suspension of fuel handling activities is directed by LFHS, then suspend fuel handling activities as follows:

1. NOTIFY the Control Room of the emergency and request the Shift Manager REVIEW EAL's.
2. **Complete assembly move or Return fuel assembly to original location.**
3. **DIS-ENGAGE fuel assembly.**
4. **RAISE grapple to "Full Up" position.**
5. DE-ENERGIZE fuel handling bridges using main circuit breaker on the bridge:
   - Main: FH-A-1-BK2
   - Aux: FH-A-2-BK2
   - SF: FH-A-3-BK2
6. TRANSFER both fuel carriages to the SF pool.

Technical Reference(s): 1505-1 (p10; Rev 55)
OP-TM-MAP-C0101 (p6, Rev 1) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 252-GLO-7 (As available)

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

Question History: 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	6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.21	
	Importance Rating		4.1

Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: SRO Question # 95

Plant Conditions:

- Reactor power is 100%, with ICS in full automatic.

Sequence of events:

DAY/TIME		EVENT
Day 1	0700 hours	EG-Y-1A and EG-Y-1B are in ES standby.
Day 1	1000 hours	EG-Y-1B is found to be inoperable due to failure of ESAS start signal relay. 7 day and 30 day timeclocks are commenced. EG-Y-1A has an IDENTICAL relay.
Day 1	1200 hours to obtain.	Replacement relays have been ordered, but will take 24 hours

Based on these conditions, identify the ONE selection below that describes the (1) T.S. required action(s) and (2) DIESEL GENERATOR related timeclock(s).

A. (1) Perform surveillance test run on EG-Y-1A by Day 2, 1000 hours.
(2) Commence 12 hour timeclock when EG-Y-1A is taken out of ES standby for testing.

B. (1) Perform surveillance test run on EG-Y-1A by Day 2, 1000 hours.
(2) Declare NO extra timeclocks when EG-Y-1A is taken out of ES standby for testing.

C. (1) Perform NO surveillance test run on EG-Y-1A until AFTER relay replacements.
(2) Commence 12 hour timeclock on Day 1, 1000 hours.

D. (1) Perform NO surveillance test run on EG-Y-1A until AFTER relay replacements.
(2) Declare NO extra timeclocks when EG-Y-1A is taken out of ES standby for testing.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect but plausible because Tech Spec 3.7.2 states:

> In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

Until the 24 hour grace period is expired or EG-Y-1B is declared to be inoperable, this Tech Spec does not apply.

B. **Correct.**

Part 1: Per Tech Spec 3.7.2.c:

> Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days provided that the redundant diesel generator is:
> 1. verified to be operable immediately;
> 2. within 24 hours, either:
>   1. determine the redundant diesel generator is not operable due to a common mode failure; or,
>   2. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

Since EG-Y-1B was discovered to be out of service at 1000 on Day 1, 24 hours later will be 1000 on Day 2.

Although the failed part is identical in both EDG's, there is no proof or any condition given that makes EG-Y-1A inoperable immediately.

Part 2: IAW Tech Spec 3.7.2.C.2.a, until the 24 hour grace period is expired or EG-Y-1A is declared to be inoperable, there are no additional timeclocks required.

Tech Spec 3.7 bases:

> It is recognized that while testing the redundant emergency diesel generator (EDG) in accordance with surveillance requirement 4.6.1.a, the EDG will not respond to an automatic initiation signal. In this situation, the 12 hour time clock will not be entered per the provisions of section 3.7.2.f. due to the low probability

of an event occurring while the EDG is being tested.

C. **Incorrect.**

Part 1 is incorrect but plausible if the candidate believes that performing the surveillance run on EG-Y-1A will cause it to become Inoperable. This train of thought would lead to belief that IAW Tech Spec 3.7.2:

> With one diesel generator inoperable, in addition to the above, verify that: All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

Part 2 is incorrect but plausible if the candidate believes that since EG-Y-1A has an identical relay, that it is considered inoperable as soon as EG-Y-1B is declared inoperable.

D. **Incorrect.**

Part 1 is incorrect but plausible if the candidate believes that performing the surveillance run on EG-Y-1A will cause it to become Inoperable. This train of thought would lead to belief that IAW Tech Spec 3.7.2:

> With one diesel generator inoperable, in addition to the above, verify that: All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

Part 2 is correct.

Technical Reference(s): Tech Spec (p3-43, 3-43a; Amend 258) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-14 (As available)

Question Source: Bank # QS-XXX-GLO-XX-Q89

Modified Bank # (Note changes or attach parent)

New

Question History: 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 2	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.11	
	Importance Rating		4.3

Ability to control radiation releases.

Proposed Question: SRO Question # 96

Plant Conditions:

- Weekly Chemistry sample results reveal:
  - The quantity of radioactivity contained in "A" Waste Gas Decay Tank (WDG-T-1A) is 10,500 curies.
- Beckman Analyzer sample results at Sample Point #1, "A" Waste Gas Decay Tank (WDG-T-1A), reveal:
  - Hydrogen at 3% by volume.
  - Oxygen at 3% by volume.

The actions required to be taken based on sample results are to immediately suspend all additions of radioactive material to the tank and ____(1)____, and the basis for the action is to ensure that a release of the contents ____(2)___.

A. (1) reduce the tank contents to within limits within 48 hours
(2) will be controlled in conformance with the requirements of 10 CFR 50 upon an explosion.

B. (1) reduce the tank contents to within limits within 48 hours
(2) will not exceed 0.5 rem total body exposure to a member of the public at the nearest exclusion boundary.

C. (1) reduce the tank contents to within limits immediately, without delay
(2) will be controlled in conformance with the requirements of 10 CFR 50 upon an explosion.

D. (1) reduce the tank contents to within limits immediately, without delay
(2) will not exceed 0.5 rem total body exposure to a member of the public at the nearest exclusion boundary.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct:

Offsite Dose Calculation Manual, Section 2.2.2.6 Waste Gas Decay Tanks
The quantity of radioactivity contained in each waste gas decay tank shall be limited to less than or equal to 8800 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times

ACTION: a. With the quantity of radioactive material in any waste gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

Part 2 is incorrect but plausible if the examinee believes that the hydrogen and oxygen concentrations are high enough to cause an explosive mixture, and confuses the timeclock with that of high radioactivity.

B. **Correct.**

Part 1 is correct:

Offsite Dose Calculation Manual, Section 2.2.2.6 Waste Gas Decay Tanks
The quantity of radioactivity contained in each waste gas decay tank shall be limited to less than or equal to 8800 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times

ACTION: a. With the quantity of radioactive material in any waste gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit. handling accident.

Part 2 is correct:

Offsite Dose Calculation Manual, Section 2.2.2.6 Waste Gas Decay Tanks

BASES: Restricting the quantity of radioactivity contained in each waste gas decay tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

C. **Incorrect.**

Part 1 is incorrect:

Offsite Dose Calculation Manual, Section 2.2.2.5 Explosive Gas Mixture

The concentration of oxygen in the Waste Gas Holdup System shall be limited to less than or equal to 2% by volume whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume.

AVAILABILITY: At all times

ACTION: Whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume, and:

- The concentration of oxygen in the Waste Gas Holdup System is greater than 2% by volume, but less than 4% by volume, without delay, begin to reduce the oxygen concentration to within its limit.

Plausible if the examinee recognizes the Oxygen limit but not the related Hydrogen limit of 4%. At 3% each, an explosive mixture does NOT exist and actions need not be taken per the ODCM.

Part 2 is incorrect but plausible if the examinee believes that the hydrogen and oxygen concentrations are high enough to cause an explosive mixture, and confuses the timeclock with that of high radioactivity.

D. **Incorrect.**

Part 1 is incorrect:

Offsite Dose Calculation Manual, Section 2.2.2.5 Explosive Gas Mixture

The concentration of oxygen in the Waste Gas Holdup System shall be limited to less than or equal to 2% by volume whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume.

AVAILABILITY: At all times

ACTION: Whenever the concentration of hydrogen in the Waste Gas Holdup System is greater than or equal to 4% by volume, and:

- The concentration of oxygen in the Waste Gas Holdup System is greater than 2% by volume, but less than 4% by volume, without delay, begin to reduce the oxygen concentration to within its limit.

Plausible if the examinee recognizes the Oxygen limit but not the related Hydrogen limit of 4%. At 3% each, an explosive mixture does NOT exist and actions need not be taken per the ODCM.

Part 2 is correct:

Offsite Dose Calculation Manual, Section 2.2.2.6 Waste Gas Decay Tanks

BASES: Restricting the quantity of radioactivity contained in each waste gas decay tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

Technical Reference(s): ODCM (p41; Rev 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 231-GLO-14 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: 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 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.27	
	Importance Rating		3.9

Knowledge of "fire in the plant" procedure.

Proposed Question: SRO Question # 97

Plant conditions:

- Reactor is in a refueling outage.
- The Fuel Transfer Canal water level is 24 ft above the Reactor Vessel Flange with the Reactor Vessel Head removed.

Event:

- A fire is in progress in the Auxiliary Building A DH Vault.
- Shift Manager has ordered you, the assigned Fire Brigade Advisor, to remain in the Control Room to address the loss of the A DH Train.

Identify the ONE selection below that describes how to (1) fulfill your FIRE BRIGADE ADVISOR responsibilities FROM THE CONTROL ROOM, in accordance with 1038, Administrative Controls – Fire Protection Program, and (2) how to address the loss of the A DH Train.

A. (1) IAW OP-TM-AOP-001-F01, Fire in Fuel Handling Building 281', assess potential operational effects of the fire.
(2) At least two means for maintaining DHR capability shall be OPERABLE or immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

B. (1) IAW OP-TM-AOP-001-F01, Fire in Fuel Handling Building 281', assess potential operational effects of the fire.
(2) No Tech Spec violation; only one DHR Loop is required to be operable because the volume of water above the Reactor Vessel flange provides a large heat sink which would allow sufficient time to recover active DHR means.

C. (1) IAW OP-TM-AOP-001, Fire, identify safe shutdown equipment at risk to be considered in the fire mitigation strategy.
(2) At least two means for maintaining DHR capability shall be OPERABLE or immediately initiate corrective action to return the required loops to OPERABLE

status as soon as possible.

D. (1) IAW OP-TM-AOP-001, Fire, identify safe shutdown equipment at risk to be considered in the fire mitigation strategy.
(2) No Tech Spec violation; only one DHR Loop is required to be operable because the volume of water above the Reactor Vessel flange provides a large heat sink which would allow sufficient time to recover active DHR means.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect.**

Part (1) is plausible because it is the location of the fire, but the procedure is only applicable if RCS temperature is >200F.

Part (2) is plausible because T.S. 3.4.2.1 states:

At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").
b. DHR String (Loop "B").
c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.
d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

However, T.S. 3.4.2.3 is applicable.

B. **Incorrect.**

Part (1) is plausible because it is the location of the fire, but the procedure is only applicable if RCS temperature is >200F.

Part (2) is correct:

T.S. 3.4.2.3 The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

C. **Incorrect.**

Part (1) is correct.

Part (2) is plausible because T.S. 3.4.2.1 states:

At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").
b. DHR String (Loop "B").
c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.
d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

However, T.S. 3.4.2.3 is applicable.

D. **Correct**:

(1) 1038, section 4.11.4:

A Control Room Operator (any licensed operator) initially responds to the fire scene for fires in safety related areas to act as a Fire Brigade Advisor (FBA). The purpose of the FBA is to assess the potential impact of the fire on plant operation and safety. The FBA works with and provides advice to the fire brigade leader on which "at-risk" plant equipment is most significant so that this information may be considered in the fire mitigation strategy. The FBA alerts the Control Room team on the potential impact of the fire so that the Control Room team may take appropriate compensatory or contingency actions. After the Control Room has been briefed by the FBA, the Shift Manager may re-assign the FBA based on other priorities. Fire brigade qualifications do not apply to this position. This position is not required to respond to TMI-2.

AOP-001 is entered when a confirmed unexpected fire or smoke is observed within TMI owner controlled area, **and** the fire has **not** been extinguished when reported.

(2) T.S. 3.4.2.3

The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

Technical Reference(s): 1038 (p10, Rev 77)
Tech Spec 3.4.2.3
OP-TM-AOP-001 (p1 Rev 8)

(Attach if not previously provided)

Proposed References to be provided to applicants during examination:			None
Learning Objective:	AOP001-PCO-1		(As available)
Question Source:	Bank #		
	Modified Bank # X		(Note changes or attach parent)
	New		
Question History:		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 2		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.7	
	Importance Rating		4.7

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: SRO Question # 98

Plant conditions:

- Reactor is subcritical, Safety Rod Groups 1-4 are fully withdrawn.
- Ready for reactor startup using 1103-8, Approach to Criticality.
- Last three samples verify RCS boron concentration is at the value required for the desired critical rod position.

Event:

- First continuous rod withdrawal (70-seconds) repositions CRD Group 5 rods from 0% to 25% withdrawn.
- NI-11 and NI-12 count rate indications respond similarly, during and after the rod motion.
- NI-11 Startup Rate Indication:
  - o During the first 5 seconds of rod motion, NI-11 startup rate indication increased to + 0.2 dpm, and then continued to ramp up to a maximum of +0.3 dpm during the rest of the rod motion.
  - o When rod motion was stopped, NI-11 startup rate lowered to +0.1 dpm in the next second, and then lowered to 0 dpm over the next minute.
- NI-12 Startup Rate Indication:
  - o During the entire period of rod motion, NI-12 startup rate indication increased linearly from 0 dpm to a maximum of + 0.2 dpm.
  - o When rod motion was stopped, it reduced linearly to 0 dpm over the 70 seconds.

Based on these conditions, identify the ONE selection below that describes:
(1) Operational condition of Source Range nuclear instrumentation systems.
(2) Appropriate actions to be implemented.

A. (1) NI-11 is NOT operable.
(2) The reactor startup is permitted to continue up to full rated power using plant procedures.

B. (1) NI-11 is NOT operable.
(2) Insert control rods to render the reactor at least 1% dK/K subcritical due to not

meeting the redundancy and single failure criteria.

C. (1) NI-12 is NOT operable.
(2) The reactor startup is permitted to continue up to full rated power using plant procedures.

D. (1) NI-12 is NOT operable.
(2) Insert control rods to render the reactor at least 1% dK/K subcritical due to not meeting the redundancy and single failure criteria.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect.**

Part 1 is incorrect. NI-11 shows normal behavior.

Part 2 is correct.

B. **Incorrect.**

Part 1 is incorrect. NI-11 shows normal behavior.

Part 2 is incorrect. Table 3.5-1 allows for a minimum of 1 channel with a redundancy of 0. This is a rare case as most instruments require a redundant component. Plausible if the candidate believes that there is a redundancy requirement for Start-Up Range Instruments.

C. **Correct.**

Part 1 is correct. Tech Spec 1.3 states:

1.3 OPERABLE

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

1.4.1 INSTRUMENT CHANNEL

An instrument channel is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An instrument channel may be either analog or digital.

NI-12 is an instrument made up of Start-Up Rate and Count Rate. If one of the two is inoperable, then the entire instrument is inoperable.

Part 2 is correct. Tech Spec 3.1.5 Bases:

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

Tech Spec Table 3.5-1, Instrument Operating Conditions:

Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System			
4. Source range instrument	1	0	(a) (c)

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours.
(c) When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, source range instrumentation is not required.

With NI-11 operable, the Minimum Operable Channel and Minimum Degree of Redundancy criteria are met.

D. **Incorrect.**

Part 1 is correct.

Part 2 is incorrect. Table 3.5-1 allows for a minimum of 1 channel with a redundancy of 0. This is a rare case as most instruments require a redundant component. Plausible if the candidate believes that there is a redundancy requirement for Start-Up Range Instruments.

Technical Reference(s): T.S (p1-2, 3-27a, 3-29,3-30; Amendment 273) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-003-PCO-5 (As available)

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

Question History: 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	2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.13	
	Importance Rating		3.8

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: SRO Question # 99

Plant Conditions:

- A Site Area Emergency has been declared.
- The Technical Support Center (TSC) and Emergency Operations Facility (EOF) are activated with command and control functions transferred accordingly.
- An emergency exposure of greater than 5 Rem TEDE is required to terminate a radioactive release.

According to EP-AA-113 "Personnel Protective Actions", who must authorize the emergency exposure?

1. The Shift Manager in the Control Room
2. The Station Emergency Director in the TSC
3. The Corporate Emergency Director in the EOF

A. 1 ONLY

B. 2 ONLY

C. Both 1 and 2

D. Both 2 and 3

Proposed Answer: B

Explanation (Optional):

A. **Incorrect.**
Since the TSC is activated, the Shift Manager (Shift Emergency Director) has

transferred this responsibility to the Station Emergency Director.

B. **Correct.**
Per EP-AA-1009 (among others), emergency exposure controls are nondelegable responsibilities that remain with the Station Emergency Director. Since the TSC is activated, the Shift Manager (Shift Emergency Director) has transferred this responsibility to the Station Emergency Director. Per EP-AA-113, the Station Emergency Director (TSC) authorizes emergency exposures greater than 5 Rem TEDE.

C. **Incorrect.**
Since the TSC is activated, the Shift Manager (Shift Emergency Director) has transferred this responsibility to the Station Emergency Director.

D. **Incorrect.**
Per EP-AA-1009 (among others), emergency exposure controls are nondelegable responsibilities that remain with the Station Emergency Director.

Technical Reference(s): EP-AA-1009; EP-AA-113 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: N-TM-TQ-104-NOP-DBIG-APCO-1 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: PB 2009

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.41	
	Importance Rating		4.6

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: SRO Question # 100

T=0 minutes:

- PPC is unavailable.
- As a consequence of a failure of all off-site and on-site AC power sources, reactor coolant temperature is 180 degrees F and rising at 1.5°F per minute.

T=10 minutes:

- Maintenance reports that power will NOT be available for at least 10 minutes.

T=15 minutes:

- The Emergency Director has made an Emergency Action Level (EAL) Declaration.

T=20 minutes:

- Both Class 1E busses now energized from Emergency Diesel Generators.

Given the above conditions:
(1) At T=15 minutes what is the EAL Declaration made by the Emergency Director, and
(2) By T=35 minutes, what action is required by the Emergency Director?

A. (1) MA2
(2) Maintain current EAL Declaration.

B. (1) MA2
(2) Downgrade the EAL Declaration.

C. (1) MS1
(2) Maintain current EAL Declaration.

D. (1) MS1
(2) Downgrade the EAL Declaration.

Proposed Answer: A

Explanation (Optional):

EP-AA-1009, EXELON NUCLEAR RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION:

Definitions:

Mode 3, Hot Shutdown: The plant is in the Hot Shutdown (HSD) condition when the reactor is subcritical by at least one percent delta k/k and Tavg is at or greater than 525°F.

Mode 4, Heatup/Cooldown: The plant is in the Heatup/Cooldown (HU/CD) condition when the reactor coolant temperature is greater than 200°F and less than 525°F.

Mode 5, Cold Shutdown: The plant is in the Cold Shutdown (CSD) Condition when the reactor is subcritical by at least one percent delta k/k and Tavg is no more than 200°F. Additionally the reactor coolant system pressure allowed is defined by Technical Specification 3.1.2.

A. **Correct answer.**

Part (1):
IAW OP-TM-1009, RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION, Section 3.2:

> If an event occurs, and a lower or higher plant-operating mode is reached before the emergency classification can be made, the declaration shall be based on the mode that existed at the time the event occurred.

Additionally, the EAL Threshold Value for MA2 has a NOTE:

> NOTE: The Emergency Director should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

At T=10 minutes, the plant is still in Mode 5, Cold Shutdown, based on reactor coolant temperature [180F + (1.5F/min)(10min)=195F]. Therefore the EAL call in Mode 5 is MA2, based on the loss of all off-site and on-site AC power sources.

MA2 Loss of All Offsite Power and Loss All Onsite AC Power to Essential Busses.

1. Loss of power to Auxiliary Transformers 1A and 1B.
AND
2. Failure of EG-Y-1A, EG-Y-1B Emergency Diesel Generators and EG-Y-4 SBO Diesel Generator to supply power to Emergency 4KV Buses.
AND

- Failure to restore power to either Emergency 4KV Bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Part (2) is correct.
By T=35 minutes, several things have changed:
1. Reactor coolant temperature has risen above 200F, and therefore the Hot Matrix is applicable for new EAL's.
2. Power has been restored to the 1D and 1E 4kV ES busses.

However, IAW OP-TM-1009, RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION, Section 3.2:

> If there is a change in Mode following the initial event declaration, any subsequent events will be evaluated on the existing Mode of the plant at the time the conditions occur.

Which means that although the plant has heated up to Mode 4, the EAL is based on MODE 5 conditions. Additionally, although the criteria for MU1 is now met, it is not possible within TMI procedures to downgrade the EAL. The only option to exit the EAL is when recovery/termination is met, and as long as off-site power is not available, recovery/termination cannot occur.

B. **Incorrect.**
Part (1) is correct.

Part (2) is incorrect:
IAW IAW OP-TM-1009, RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION, a change of Mode does not change the declaration.

Plausible if the candidate believes that with the restoration of an on-site AC source, the EAL can be downgraded to MU1, whether in the Cold Matrix or the Hot Matrix.

C. **Incorrect.**
Part (1) is incorrect but plausible if the candidate incorrectly uses the Hot Matrix.

Part (2) is correct.

D. **Incorrect.**
Part (1) is incorrect but plausible if the candidate incorrectly uses the Hot Matrix.

Part (2) is incorrect:
IAW IAW OP-TM-1009, RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION, a change of Mode does not change the declaration.

Plausible if the candidate believes that with the restoration of an on-site AC source, the EAL can be downgraded to MU1, whether in the Cold Matrix or the Hot Matrix.

Technical Reference(s): EP-AA-1009 (p3-23,3-31; Rev 17) (Attach if not previously provided)

Proposed References to be provided to applicants during examination:		EP-AA-1009, pages 3-11 through 3-29
Learning Objective:		(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	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 5	

Comments: