SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

239002 SRVs

K1.05 - Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Plant air systems: Plant-Specific

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	239002	
Importance Rating	3.1	

Proposed Question: **# 1**

Regarding the pneumatic supply for the pressure relief function and the ADS function of the Safety Relief Valves (SRVs) which one of the following completes the statement below?

Both functions are supplied by the ________(1)__________ system and _____(2)_____ backup supply bottles.

	(1)	(2)
A.	Instrument Air	both functions have
B.	Instrument Air	only the ADS function has
C.	Containment Instrument Gas	both functions have
D.	Containment Instrument Gas	only the ADS function has

Proposed Answer:	**D**

Explanation:

a. Incorrect – SRVs are supplied by Containment Instrument Gas, not Instrument Air. The pressure relief function does not have backup supply bottles.

b. Incorrect - SRVs are supplied by Containment Instrument Gas, not Instrument Air.

c. Incorrect - The pressure relief function does not have backup supply bottles.

d. Correct – Both the pressure relief and ADS function of the SRVs is supplied nitrogen from by Containment Instrument Gas. The pressure relief function is supplied by the 90 psig header, which is not provided with backup supply bottles. The ADS function is supplied by the 150 psig header, which is provided with backup supply bottles.

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Technical Reference(s): TM-OP-025 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215003 IRM	Tier #	2	
K1.01 - Knowledge of the physical connections and/or cause-effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: RPS	Group #	1	
	K/A #	215003	
	Importance Rating	3.9	

Proposed Question: **# 2**

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

- All IRMs are on range 8.
- IRMs are indicating as follows:

IRM A	105
IRM B	117
IRM C	103
IRM D	98
IRM E	107
IRM F	111
IRM G	122
IRM H	101

Which one of the following describes the resulting status of the Reactor Protection System (RPS)?

A. Neither side of RPS receives a half scram.

B. RPS A receives a half scram, but RPS B does NOT.

C. RPS B receives a half scram, but RPS A does NOT.

D. A full scram is received.

Proposed Answer:	**B**

Explanation:

a. Incorrect – RPS A receives a half scram because IRM G has exceeded the scram setpoint of 120.

b. Correct – RPS A receives inputs from IRMs A, C, E, and G. RPS B receives inputs from IRMs B, D, F, and H. A single IRM reading above the scram setpoint of 120 will cause a half scram on the respective side of RPS. RPS A receives a half scram because IRM G has exceeded the scram setpoint of 120. No half scram is received on RPS B because none of its IRMs have exceeded 120. IRM B is above the upscale alarm of 108, which causes a rod block, but not a half scram.

c. Incorrect - RPS A receives a half scram because IRM G has exceeded the scram setpoint of 120. No half scram is received on RPS B because none of its IRMs have exceeded 120.

d. Incorrect - No half scram is received on RPS B because none of its IRMs have exceeded 120.

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Technical Reference(s): AR-104-A05, AR-104-A06 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 NRC #36 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:

203000 RHR/LPCI: Injection Mode

K2.02 - Knowledge of electrical power supplies to the following: Valves

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	203000	
Importance Rating	2.5	

Proposed Question: **# 3**

Unit 1 is operating at 100% power when 480V Isolation Swing Bus 1B219 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on the ability to **ALIGN** RHR loop 1A for LPCI and Drywell spray from the Control Room?

A flow path for RHR loop 1A...

A. can NOT be established from the Control Room for either LPCI or Drywell spray.

B. can be established from the Control Room for LPCI, but NOT Drywell spray.

C. can be established from the Control Room for Drywell spray, but NOT LPCI.

D. can be established from the Control Room for both LPCI and Drywell spray.

Proposed Answer:	C

Explanation:

a. Incorrect - The Drywell spray isolation valves (HV-151-F016A and -F021A) are powered from 1B217 and are unaffected by the loss of 1B219. Therefore, Drywell spray can still be aligned on RHR loop 1A from the Control Room.

b. Incorrect - 1B219 supplies power to the normally closed LPCI injection valve HV-151-F015A. With no power on 1B219, this valve cannot be opened from the Control Room, therefore LPCI cannot be aligned from the Control Room (nor will it automatically align). The Drywell spray isolation valves (HV-151-F016A and -F021A) are powered from 1B217 and are unaffected by the loss of 1B219. Therefore, Drywell spray can still be aligned on RHR loop 1A from the Control Room.

c. Correct – 1B219 supplies power to the normally closed LPCI injection valve HV-151-F015A. With no power on 1B219, this valve cannot be opened from the Control Room, therefore LPCI cannot be aligned from the Control Room (nor will it automatically align). The Drywell spray isolation valves (HV-151-F016A and -F021A) are powered from 1B217 and are unaffected by the loss of 1B219. Therefore, Drywell spray can still be aligned on RHR loop 1A from the Control Room.

d. Incorrect - 1B219 supplies power to the normally closed LPCI injection valve HV-151-F015A. With no power on 1B219, this valve cannot be opened from the Control Room, therefore LPCI cannot be aligned from the Control Room (nor will it automatically align).

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-049 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # JAF 4/14 NRC #3

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam JAF 4/14 NRC #3

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

400000 Component Cooling Water

K2.01 - Knowledge of electrical power supplies to the following: CCW pumps

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	400000	
Importance Rating	2.9	

Proposed Question: **# 4**

Unit 1 is operating at 100% power with the following:

- RBCCW pump 1B is running.
- RBCCW pump 1A is in standby.
- Then, the following sequence of events occurs:

Time (minutes)	Event
0	Power is simultaneously lost to Motor Control Centers (MCCs) 1B216 and 1B237.
2	Power is restored to MCC 1B237.
4	Power is restored to MCC 1B216.

Which one of the following describes the resulting operation of the RBCCW pumps at time 5 minutes?

A. Both RBCCW pumps are running.

B. Neither RBCCW pump is running.

C. RBCCW pump 1A is running and RBCCW pump 1B is NOT running.

D. RBCCW pump 1B is running and RBCCW pump 1A is NOT running.

Proposed Answer: **D**

Explanation:

a. Incorrect – RBCCW pump 1A would have started at time 4 minutes if a low pressure signal still existed, however RBCCW pump 1B already started and restored system pressure at time 2 minutes.

b. Incorrect – RBCCW pumps are provided with automatic start logic on low system pressure. Therefore, when power is restored to the first pump (pump 1B at time 2 minutes), it automatically starts and then remains running unless operator action is taken.

c. Incorrect – MCC 1B237 is restored first and it supplies power to RBCCW pump 1B, not 1A.

d. Correct – RBCCW pump 1A is powered by MCC 1B216 and RBCCW pump 1B is powered by MCC 1B237. At time 0 minutes, both pumps will be secured due to lack of power. At time 2 minutes, RBCCW pump 1B will automatically re-start due to restoration of MCC 1B216 and a low pressure signal. RBCCW pump 1B will restore system flow and pressure such that at time 4 minutes, RBCCW pump 1A will NOT start on a low pressure signal when power is restored to MCC 1B216.

Technical Reference(s): TM-OP-014 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 NRC #53

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC24 NRC #53

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

209001 LPCS

K3.03 - Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: Emergency generators

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	209001	
Importance Rating	2.9	

Proposed Question: **# 5**

Unit 1 has experienced a loss of offsite power (LOOP) and a loss of coolant accident (LOCA) with the following:

- Emergency Diesel Generators (EDGs) A, B, C, and D are powering their respective buses.
- B Loop Core Spray is injecting approximately 3950 gpm to the Reactor.
- HV-152-F005B, CORE SPRAY LOOP B IB INJ SHUTOFF, is mid-position.
- Then, a malfunction causes HV-152-F005B to fully open.

Which one of the following describes the change in EDG loading due to this malfunction?

A. EDG B loading rises. Loading on EDGs A, C, and D remains stable.

B. EDG B and C loading rises. Loading on EDGs A and D remains stable.

C. EDG B and D loading rises. Loading on EDGs A and C remains stable.

D. EDG C and D loading rises. Loading on EDGs A and B remains stable.

Proposed Answer:	C

Explanation:

a. Incorrect – While EDG B loading rises, EDG D loading also rises because HV-152-F005B also controls flow from Core Spray pump 1D.

b. Incorrect – While EDG B loading rises, EDG C load is unaffected. EDG C powers only Core Spray pump 1C, which is in Core Spray loop A, not B. Additionally, EDG D loading also rises because HV-152-F005B also controls flow from Core Spray pump 1D.

c. Correct – HV-152-F005B controls flow from both Core Spray pumps 1B and 1D. With a loss of offsite power, these pumps are powered by EDGs B and D, respectively. Core Spray loop 1B is currently injecting approximately half of the rated emergency flow. When HV-152-F005B fully opens, flow from Core Spray pumps 1B and 1D rises, causing pump current draw to rise. This causes loading on EDGs B and D to rise. EDGs A and C are unaffected.

d. Incorrect – While EDG D loading rises, EDG C load is unaffected. EDG C powers only Core Spray pump 1C, which is in Core Spray loop A, not B. Additionally, EDG B loading also rises because HV-152-F005B also controls flow from Core Spray pump 1B.

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Technical Reference(s): OP-151-001, TM-OP-151 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Examination Outline Cross-reference:

211000 SLC

K3.02 - Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: Core spray line break detection system: Plant-Specific

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	211000	
Importance Rating	3.0	

Proposed Question: **# 6**

Unit 1 is operating at 100% power when the Standby Liquid Control injection line develops a significant leak inside the Drywell.

Which one of the following describes the potential impact of this leak?

This leak could result in...

A. an incorrect indication of Core Plate D/P.

B. an incorrect indication of a Core Spray line break.

C. an incorrect indication of Reactor head seal leakage.

D. an incorrect indication of Control Rod Drive Cooling D/P.

Proposed Answer:	**A**

Explanation:

a. Correct - The SLC injection line connects to the inner pipe of the "pipe-within-a-pipe" and serves as the high pressure side of the core plate D/P transmitter input. The outer pipe of the pipe-within-a-pipe surrounds the SLC injection sparger, continues through the core plate, and serves as the lower pressure side of the core plate D/P transmitter input. A significant leak in the SLC injection line would lower the sensed pressure on the high pressure side of the Core Plate D/P transmitter and result in an incorrect D/P reading.

b. Incorrect - The Core Spray line break detection system uses a pressure from the outer pipe of the pipe-within-a-pipe, however the inner pipe (which is connected to the SLC injection line) does not affect the Core Spray line break detection system.

c. Incorrect - The Reactor head seal leakage system uses a stand-alone pressure tap to detect build-up of pressure between the Reactor head seals, and is therefore unaffected by the SLC injection line leak.

d. Incorrect – The CRD cooling water D/P transmitter uses a pressure from the outer pipe of the pipe-within-a-pipe, however the inner pipe (which is connected to the SLC injection line) does not affect the CRD cooling water D/P indication.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-053 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP1 2015 NRC #1

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam NMP1 2015 NRC #1

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
215005 APRM / LPRM	Tier #	2	
K4.08 - Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Sampling of overall core power in each APRM (accomplished through LPRM assignments and symmetrical rod patterns)	Group #	1	
	K/A #	215005	
	Importance Rating	2.7	

Proposed Question: **# 7**

Which one of the following completes the below statement?

The LPRMs assigned to APRM 1 cover ___(1)___ quadrants and ___(2)___ detector levels?

	(1) Number of Core Quadrants	(2) Number of Detector Levels
A.	1	2
B.	1	4
C.	4	2
D.	4	4

Proposed Answer:	D

Explanation:

a. Incorrect – APRM 1 receives inputs from LPRMs in each of the four core quadrants, not just one. APRM 1 also receives inputs from LPRMs at each of the four detector levels, not just two.

b. Incorrect – APRM 1 receives inputs from LPRMs in each of the four core quadrants, not just one.

c. Incorrect – APRM 1 receives inputs from LPRMs at each of the four detector levels, not just two.

d. Correct – APRM 1 receives inputs from LPRMs in each of the four core quadrants and at each of the four detector levels.

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Technical Reference(s): OP-178-002 Attachment B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor	Tier #	2	
K4.01 - Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks	Group #	1	
	K/A #	215004	
	Importance Rating	3.7	

Proposed Question: **# 8**

A Unit 1 startup is in progress with the following:

- A control rod withdrawal block is received.
- The following neutron monitoring indications exist:
  - SRMs A and B: Fully inserted and indicate 8×10^4 cps
  - SRMs C and D: Partially withdrawn and indicate 60 cps
  - Reactor period: Indicates +200 seconds
  - IRMs: Fully inserted and indicate 4 on Range 1

Which one of the following will clear the control rod withdrawal block?

A. Drive in SRMs C and D until they indicate greater than 100 cps.

B. Drive out SRMs A and B until they indicate less than 1×10^4 cps.

C. Insert a control rod until Reactor period is longer than +300 seconds.

D. Allow power to rise until the IRMs indicate greater than 5 on range 1.

Proposed Answer: **A**

Explanation:

a. Correct – The rod block is being caused by SRMs C and D indicating less than 100 cps while not fully inserted and IRMs below range 3. Therefore, the rod block will clear if SRMs C and D are inserted until they indicate greater than 100 cps.

b. Incorrect – The rod block is not being caused by upscale SRMs A and B because they are below the upscale setpoint of 1 x 10^5 cps.

c. Incorrect – The rod block is not being caused by Reactor period because it is longer than the +50 second setpoint.

d. Incorrect – The rod block is not being caused by IRM downscale because this rod block is bypassed with the IRMs on range 1.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): AR-104-H03, AR-104-E06, TM-OP-078A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC23 NRC #9

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC23 NRC #9

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

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REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
212000 RPS	Tier #	2	
K5.02 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements	Group #	1	
	K/A #	212000	
	Importance Rating	3.3	

Proposed Question: **# 9**

Unit 1 is operating at 100% power with the Div 2 RPS system de-energized.

Which one of the following describes the MINIMUM action required to cause a full Reactor scram?

Arm and depress...

A. EITHER the A1 OR the A2 pushbutton.

B. BOTH the A1 AND the A2 pushbuttons.

C. EITHER the A1 OR the A2 pushbutton, AND EITHER the B1 OR B2 pushbutton.

D. ALL FOUR of the A1, B1, A2, and B2 pushbuttons.

Proposed Answer:	A

Explanation:

a. Correct – There is already a half scram on RPS B due to Div 2 RPS being de-energized. Either the A1 or the A2 pushbutton being armed and depressed will satisfy the logic to cause a half scram on RPS A, and therefore a full Reactor scram.

b. Incorrect – Only one of these two pushbuttons needs to be armed and depressed to complete the logic for a half scram on RPS A, therefore this is not the minimum action required.

c. Incorrect – Neither of the B pushbuttons need to be depressed, since the RPS logic is arranged such that Div 2 RPS de-energizing causes a half scram on RPS B.

d. Incorrect – Neither of the B pushbuttons need to be depressed, since the RPS logic is arranged such that Div 2 RPS de-energizing causes a half scram on RPS B. Only one of the two A pushbuttons needs to be armed and depressed to complete the logic for a half scram on RPS A.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-058 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 Cert #8

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
264000 EDGs	Tier #	2	
K5.06 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Load sequencing	Group #	1	
	K/A #	264000	
	Importance Rating	3.4	

Proposed Question: **# 10**

Unit 1 is operating at 100% power with the following:

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

Which one of the following describes the approximate times for the starting sequence of the low pressure ECCS pumps after the EDG output breakers close?

	RHR A & B	RHR C & D	Core Spray A, B, C, & D
A.	0 seconds	3 seconds	10.5 seconds
B.	0 seconds	7 seconds	15.0 seconds
C.	3 seconds	3 seconds	10.5 seconds
D.	3 seconds	7 seconds	15.0 seconds

Proposed Answer:	C

Explanation:

a. Incorrect – RHR pumps A and B start after a 3 second time delay, not immediately after EDG output breaker closure (which is the loading scheme if offsite power is available).

b. Incorrect – RHR pumps A and B start after a 3 second time delay, not immediately after EDG output breaker closure. RHR pumps C and D start after a 3 second time delay, not 7 seconds. Core Spray pumps start after a 10 second time delay, not 15 seconds. This answer represents the loading scheme if offsite power were available.

c. Correct – With a simultaneous LOOP/LOCA, once the EDG output breakers close, all RHR pumps start after a 3 second delay and all Core Spray pumps start after a 10 second delay.

d. Incorrect – RHR pumps C and D start after a 3 second time delay, not 7 seconds. Core Spray pumps start after a 10 second time delay, not 15 seconds. These delays are based on the loading scheme if offsite power were available.

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Technical Reference(s): TM-OP-49, TM-OP-51 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP2 2014 NRC #7

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam NMP2 2014 NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

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REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
218000 ADS	Tier #	2	
K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: RHR/LPCI system pressure: Plant-Specific	Group #	1	
	K/A #	218000	
	Importance Rating	3.9	

Proposed Question: **# 11**

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -130", down slow.
- Reactor pressure is 800 psig, down slow.
- Drywell pressure is 10 psig, up slow.
- RHR pump 1A is running.
- NO other ECCS pumps are running.
- NO high pressure injection sources are available.
- A leak on RHR pump 1A is resulting in a discharge pressure of 100 psig, steady.

Which one of the following describes the resulting status of the ADS valves two minutes later?

Two minutes later, the ADS valves will be __(1)__ because the ADS logic senses __(2)__.

	(1)	(2)
A.	open	adequate ECCS discharge pressure
B.	closed	inadequate ECCS discharge pressure
C.	open	an adequate number of closed ECCS pump breakers
D.	closed	an inadequate number of closed ECCS pump breakers

Proposed Answer:	**B**

Explanation:

a. Incorrect – The ADS valves will not open after the 102 second timer expires because of inadequate running ECCS pumps, as sensed by discharge pressures. One properly operating RHR pump would satisfy the logic, but RHR pump 1A discharge pressure is too low.

b. Correct – The 102 second ADS timer is counting down because Reactor water level is <-129" and Drywell pressure is >1.72 psig. The ADS valves will not open after the 102 second timer expires because an inadequate number of ECCS pumps are available, as sensed by discharge pressures. For ADS to open the valves, the logic needs to sense either both Core Spray pumps in one loop (ex. 1A and 1C) or one RHR pump operating as indicated by each of their individual discharge pressures (>135 psig for Core Spray, >121 psig for RHR). Since RHR pump 1A is the only pump running and only producing a discharge pressure of 100 psig, the logic is not satisfied and ADS valves will not open.

c. Incorrect – The ADS valves will not open after the 102 second timer expires because of inadequate running ECCS pumps, as sensed by discharge pressures.

d. Incorrect – The ADS valves will be closed, however the ADS logic senses ECCS discharge pressures, not breaker positions.

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Technical Reference(s): TM-OP-83E (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New x

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 8

55.43 ____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
263000 DC Electrical Distribution	Tier #	2	
K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION: A.C. electrical distribution	Group #	1	
	K/A #	263000	
	Importance Rating	3.2	

Proposed Question: **# 12**

Unit 1 is in Mode 3 and Unit 2 is in Mode 1 when 0B516, Diesel Generator A ESS 480V MCC, de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on DC electrical distribution?

A. Unit 1 is NOT affected.
Unit 2 Battery Charger 2D613 transfers to its alternate AC source.

B. Unit 1 Battery Charger 1D613 de-energizes and Battery 1D610 carries the DC loads.
Unit 2 is NOT affected.

C. Unit 1 Battery Charger 1D613 de-energizes and Battery 1D610 carries the DC loads.
Unit 2 Battery Charger 2D613 de-energizes and Battery 2D610 carries the DC loads.

D. Unit 1 Battery Charger 1D613 transfers to its alternate AC source.
Unit 2 Battery Charger 2D613 transfers to its alternate AC source.

Proposed Answer:	**C**

Explanation:

a. Incorrect - Both Unit 1 and 2 Battery Chargers are affected. There is no alternate source of AC power for these Battery Chargers. Portable Battery Charger 0D101 must be manually aligned to provide an alternate power to batteries.

b. Incorrect – Unit 2 is also affected.

c. Correct – 0B516 is the AC supply to both of the given Unit 1 and Unit 2 Battery Chargers. 0B516 is powered through an automatic transfer switch that ensures continuity of power upon loss of the normal supply to 0B516. However this will not result in continuity of power in this case since there is a fault on 0B516. These Battery Chargers do not have an automatic alternate AC supply, therefore the Battery Chargers de-energize and the associated batteries carry the DC loads. Portable Battery Charger 0D101 can be manually aligned to provide an alternate AC power to the batteries.

d. Incorrect – Both Units are affected, however these Battery Chargers do not have an automatic alternate AC supply, therefore they de-energize and the associated batteries carry the DC loads.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-002, ON-104-201 Attachment E (Attach if not previously provided)
ON-4KV-101

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC23 NRC #11

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC23 NRC #11

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
217000 RCIC	Tier #	2	
A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: Suppression pool level	Group #	1	
	K/A #	217000	
	Importance Rating	3.3	

Proposed Question: **# 13**

Unit 1 has experienced a Reactor scram with the following:

- Reactor water level is +30”, down slow.
- Reactor pressure is 940 psig, steady, on Turbine Bypass Valves.
- RCIC was manually started during the transient to control Reactor water level.
- The RCIC flow controller is now in MANUAL with pump discharge pressure at 900 psig.
- All Condensate and Feedwater pumps are tripped.
- HPCI, RHR, and Core Spray are all in standby.
- Suppression Pool water level is 23’.
- Condensate Storage Tank (CST) water level is 50”.

Which one of the following describes the resulting trend in Suppression Pool and CST water levels due to the current operating status of RCIC?

	Suppression Pool Water Level	CST Water Level
A.	Steady	Steady
B.	Rising	Lowering
C.	Lowering	Rising
D.	Lowering	Steady

Proposed Answer:	**B**

Explanation:

a. Incorrect – RCIC is taking suction from the CSTs and pumping minimum flow to the SP. Therefore, CST level is lowering and SP level is rising. This choice would be correct if CST level was <36", because RCIC would be pumping from the SP to the SP.

b. Correct – RCIC takes suction from the CSTs unless CST water level lowers to 36". With the flow controller in AUTO and set to 0 gpm, RCIC is on minimum flow. Therefore, about 75 gpm is being drawn from the CSTs and routed to the SP. This causes CST water level to lower and SP water level to rise.

c. Incorrect - RCIC is taking suction from the CSTs and pumping minimum flow to the SP. Therefore, CST level is lowering and SP level is rising. This choice would result if the candidate believe suction was from the SP and min flow was routed to the CSTs.

d. Incorrect - RCIC is taking suction from the CSTs and pumping minimum flow to the SP. Therefore, CST level is lowering and SP level is rising. This choice would be correct if CST level was <36" and the flow controller was set at a higher flow rate, because RCIC would be pumping from the SP to the Reactor.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-50 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # Vision SYSID 1058 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
205000 Shutdown Cooling	Tier #	2	
A1.09 - Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: SDC/RHR pump/system discharge pressure	Group #	1	
	K/A #	205000	
	Importance Rating	2.8	

Proposed Question: **# 14**

A Unit 1 shutdown is in progress with the following:

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

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

A. 420 psig

B. 300 psig

C. 204 psig

D. 95 psig

Proposed Answer:	**D**

Explanation:

a. Incorrect – The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is 98 psig due to interlocks designed to prevent over-pressurization of RHR suction piping. 420 psig is the highest Reactor pressure at which RHR LPCI injection valves open to allow injection to the Reactor in LPCI mode.

b. Incorrect – The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is 98 psig due to interlocks designed to prevent over-pressurization of RHR suction piping. 300 psig is the highest Reactor pressure at which RHR pumps are capable of injecting to the Reactor in LPCI mode.

c. Incorrect – The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is 98 psig due to interlocks designed to prevent over-pressurization of RHR suction piping. 204 psig is the pressure associated with RHR pump flow requirements.

d. Correct – The highest Reactor pressure at which RHR can inject to the Reactor in the SDC mode is 98 psig due to interlocks designed to prevent over-pressurization of RHR suction piping.

Note: The K/A requires predicting and/or monitoring changes in RHR pump discharge pressure or system discharge pressure associated with operating SDC. RHR pump discharge pressure is not a parameter specifically monitored or controlled in SDC operation. The question was not focused on RHR pump discharge pressure since the resulting question would have therefore been generic in nature and significantly overlapped with various GFE concepts. RHR system discharge pressure during SDC operation is equivalent to Reactor pressure. Reactor pressure is an important parameter that specifically impacts SDC operation. Therefore, the question meets the K/A by asking a question regarding Reactor pressure relevant to SDC operation.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-149-002 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control	Tier #	2	
A2.03 - Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of reactor water level input	Group #	1	
	K/A #	259002	
	Importance Rating	3.6	

Proposed Question: **# 15**

Unit 1 is operating at 100% power with the following:

- The 1C004 instrument rack develops a leak on the common Narrow Range Level (NRL) variable leg.
- All NRL indications on the 1C004 rack begin drifting down slow.
- Feedwater Level Control (FWLC) marks NRL channel B as DEVIANT.

Which one of the following identifies the response of Reactor water level and the operator action to be taken, in accordance with ONLVL-101, RPV Level Control Malfunction?

	Reactor Water Level	Operator Action
A.	Remains steady	Place 1C004 in Maintenance Bypass.
B.	Remains steady	Place NRL channel B in Maintenance Bypass.
C.	Rises	Attempt to stabilize Reactor water level by placing FWLC in Manual.
D.	Rises	Insert a manual Reactor scram. Manual control of FWLC is unavailable due to the malfunction.

Proposed Answer:	C

Explanation:

a. Incorrect – Reactor water level rises because FWLC has selected the two lowering NRL inputs (A & C) as valid.

b. Incorrect – Reactor water level rises because FWLC has selected the two lowering NRL inputs (A & C) as valid.

c. Correct - A variable leg leak on the 1C004 instrument panel results in slowly lowering Reactor water level indications on the N004A and C inputs to ICS and the N024A and B inputs to RPS, among others. With ICS marking the unaffected NRL input N004B as DEVIANT, the ICS level selection logic is taking the A and C inputs as indicated Reactor water level. As these indications are drifting lower, ICS begins raising FW flow to attempt to raise level. With no feedback due to the instrument drift, actual Reactor water level continues to rise. ON-LVL-101 directs placing FWLC in Manual and stabilizing Reactor water level. Since the level indications are drifting slowly, adequate time exists to take this action, rather than proceeding directly to a manual Reactor scram.

d. Incorrect – Reactor water level does rise and a manual Reactor scram may become required if operator action is unable to stabilize level. However, manual control of FWLC is available despite the failure of multiple NRL inputs. The slow downward failure of NRL inputs provides adequate time for placing FWLC in Manual.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-LVL-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC26 NRC #62 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

223002 PCIS/Nuclear Steam Supply Shutoff

A2.07 - Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Various process instrumentation failures

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002	
Importance Rating	2.7	

Proposed Question: **# 16**

Unit 1 is operating at 100% power with the following:

- At time 0 minutes, RWCU flow instrument PDIS-G33-1N044A fails upscale.
- At time 2 minutes, RWCU flow instrument PDIS-G33-1N044B fails upscale.
- At time 4 minutes, an Operator is directed to verify proper response of RWCU in accordance with AR-101-F01, RWCU HI FLOW ISOLATION.

Which one of the following describes:

(1) the **earliest approximate time** a RWCU isolation occurs, and

(2) the required isolation valve verification **at time 4 minutes**, in accordance with AR-101-F01?

A. (1) 0 minutes + 5 seconds
(2) Verify closed HV-144-F001, RWCU INLET IB ISO, only.

B. (1) 2 minutes + 5 seconds
(2) Verify closed HV-144-F001, RWCU INLET IB ISO, only.

C. (1) 0 minutes + 5 seconds
(2) Verify closed HV-144-F001, RWCU INLET IB ISO, and HV-144-F004, RWCU INLET OB ISO.

D. (1) 2 minutes + 5 seconds
(2) Verify closed HV-144-F001, RWCU INLET IB ISO, and HV-144-F004, RWCU INLET OB ISO.

Proposed Answer: **C**

Explanation:

a. Incorrect – The first isolation does occur at approximately 0 minutes + 5 seconds and does only close HV-144-F001. However by time 4 minutes, both HV-144-F001 and HV-144-F004 are required to be closed due to the second flow instrument failure.

b. Incorrect – The first isolation occurs at approximately 0 minutes + 5 seconds. By time 4 minutes, both HV-144-F001 and HV-144-F004 are required to be closed due to the second flow instrument failure.

c. Correct – Only one of the two RWCU high flow signals are required to cause a RWCU isolation. At time 0 minutes (plus nominal 5 second time delay), PDIS-G33-1N044A causes HV-144-F001 to close, which isolates RWCU. At time 2 minutes (plus nominal 5 second time delay), PDIS-G33-1N044B causes HV-144-F004 to close. Therefore at time 4 minutes, both HV-144-F001 and HV-144-F004 must be verified closed by the Operator executing AR-101-F01.

d. Incorrect – The first isolation occurs at approximately 0 minutes + 5 seconds.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): AR-101-F01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

262001 AC Electrical Distribution

A3.02 - Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Automatic bus transfer

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262001	
Importance Rating	3.2	

Proposed Question: **# 17**

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

- ESS Transformer T-111 (0X211) is out of service for planned maintenance.
- Then, the ESS Transformer T-101 (0X201) lockout relay trips.

Which one of the following describes the electrical system response?

A. Diesel Generator A starts and re-energizes ESS Buses 1A and 2A. Load Centers 1B210 and 2B210 are re-energized automatically.

B. An alternate ESS transformer re-energizes ESS Buses 1A and 2A. Load Centers 1B210 and 2B210 are re-energized automatically.

C. Diesel Generator A starts and re-energizes ESS Buses 1A and 2A. The feeder breakers to Load Centers 1B210 and 2B210 open and must be manually re-closed.

D. An alternate ESS transformer re-energizes ESS Buses 1A and 2A. The feeder breakers to Load Centers 1B210 and 2B210 open and must be manually re-closed.

Proposed Answer:	B

Explanation:

a. Incorrect – An automatic transfer scheme switches ESS Buses 1A and 2A to an alternate offsite power supply. DG A would only start if this automatic transfer scheme also failed.

b. Correct – The lockout of ESS Transformer T-101 causes undervoltage on ESS Buses 1A and 2A. Undervoltage on these buses initiates an automatic transfer scheme that re-energizes both buses from an alternate offsite power supply (T-201). Undervoltage also causes shedding of multiple Load Centers, however Load Centers 1B210 and 2B210 are exceptions that remain connected to ESS Buses 1A and 2A. Therefore, Load Centers 1B210 and 2B210 automatically re-energize.

c. Incorrect – An automatic transfer scheme switches ESS Buses 1A and 2A to an alternate offsite power supply. DG A would only start if this automatic transfer scheme also failed. Load Centers 1B210 and 2B210 are exceptions that remain connected to ESS Buses 1A and 2A. Therefore, Load Centers 1B210 and 2B210 automatically re-energize.

d. Incorrect - Load Centers 1B210 and 2B210 are exceptions that remain connected to ESS Buses 1A and 2A. Therefore, Load Centers 1B210 and 2B210 automatically re-energize.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-104-201 ON-4KV-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #7

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
261000 SGTS	Tier #	2	
A3.02 - Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: Fan start	Group #	1	
	K/A #	261000	
	Importance Rating	3.2	

Proposed Question: **# 18**

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

- A loss of all offsite power occurs.
- Both Reactors scram.
- Unit 1 Reactor water level reaches a low of -40".
- Unit 2 Reactor water level reaches a low of -30".
- Diesel Generator C fails to start.

Which one of the following describes the status of the Standby Gas Treatment (SBGT) fans?

A. Both SBGT fans are running.

B. NEITHER SBGT fan is running.

C. SBGT fan A is running and SBGT fan B is NOT running.

D. SBGT fan B is running and SBGT fan A is NOT running.

Proposed Answer:	**D**

Explanation:

a. Incorrect – Both SBGT fans would normally be running due to Unit 1 Reactor water level <-38", however the LOOP and failure of DG C result in power being unavailable to SBGT fan A.

b. Incorrect – SBGT fan B is running due to Unit 1 Reactor water level <-38".

c. Incorrect – SBGT fan A is not running due to the LOOP and failure of DG C. SBGT fan B is running due to Unit 1 Reactor water level <-38".

d. Correct – Unit 1 Reactor water level <-38" provides a signal for both SBGT fans to automatically start. The combined LOOP and DG C failure result in a loss of power to MCC 0B136, which supplies power to SBGT fan A. Therefore, only SBGT fan B is running.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-159-002 Attachment B, ON-104-203 Attachment E ON-4KV-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # Vision SYSID 33928

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 9

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

206000 HPCI

A4.13 - Ability to manually operate and/or monitor in the control room: Turbine reset control: BWR-2,3,4

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	206000	
Importance Rating	4.1	

Proposed Question: **# 19**

A transient on Unit 1 results in the following:

- HPCI automatically starts.
- Then, HPCI trips on high Reactor water level.
- Drywell pressure is 4 psig, up slow.
- Reactor water level is +55", down slow.

Which one of the following describes the response of HPCI when Reactor water level lowers to +36"?

A. FV-15612, HPCI TURB STOP VLV, automatically re-opens to re-start HPCI.

B. HV-155-F001, HPCI TURB STEAM SUPPLY VLV, automatically re-opens to re-start HPCI.

C. HPCI remains tripped. HPCI may be re-started by depressing HS-E41-1S25, HPCI HI WTR LVL TRIP RESET, which will cause FV-15612, HPCI TURB STOP VLV, to re-open.

D. HPCI remains tripped. HPCI may be re-started by depressing HS-E41-1S25, HPCI HI WTR LVL TRIP RESET, which will cause HV-155-F001, HPCI TURB STEAM SUPPLY VLV, to re-open.

Proposed Answer:	C

Explanation:

a. Incorrect - FV-15612 would only automatically re-open and re-start HPCI if Reactor water level lowered much further to -38".

b. Incorrect – HPCI remains tripped unless Reactor water level lowers much further to -38". FV-15612, not HV-155-F001, is what re-opens to re-start HPCI.

c. Correct – When Reactor water level lowers to the high level trip reset point (approximately 54"), the high level trip will remain sealed in even though a high Drywell pressure initiation signal is present. HPCI would only automatically restart if Reactor water level lowered much further to -38". However, HPCI can be manually restarted by depressing HS-E41-1S25, which causes FV-15612 to re-open.

d. Incorrect - FV-15612, not HV-155-F001, is what re-opens to re-start HPCI. HV-155-F001 opens on the initial HPCI start and remains open with HPCI tripped.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-152-001, TM-OP-52 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 Cert #1 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

300000 Instrument Air

A4.01 - Ability to manually operate and/or monitor in the control room: Pressure gauges

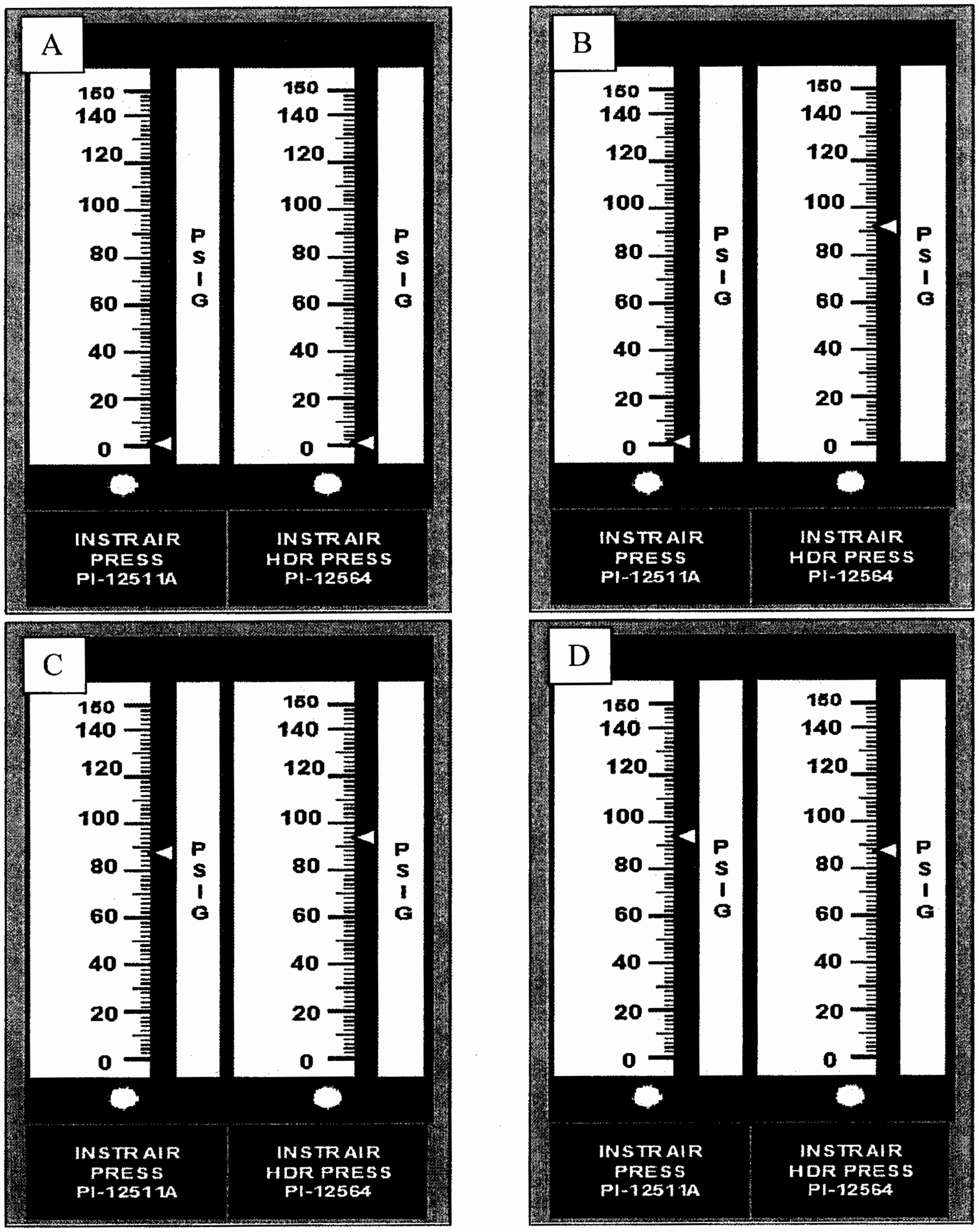
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	300000	
Importance Rating	2.6	

Proposed Question: **# 20**

Unit 1 is operating at 100% power with the following:

- Instrument Air compressor (IAC) 1A is operating as the lead compressor.
- IAC 1A is cycling between 50% and 100% loading every 5 minutes due to normal air usage.
- Instrument air is in a normal lineup with the exception of IAC 1B out of service for maintenance.
- Then, IAC 1A trips due to low oil pressure.
- Assume no operator action is taken.

Which of the following Control Room instrument air pressure indications would be expected 1 hour after the trip of IAC 1A



Proposed Answer:	**D**

Explanation:

a. Incorrect – This answer would be selected if a candidate does not recognize the normal alignment of the service air cross-connect.

b. Incorrect – This answer would be selected if a candidate assumes that service air cross-connects downstream of the dryers. (Since the IA cross tie from Unit 2 to Unit 1 is downstream of the Instrument Air dryer skids.)

c. Incorrect – This answer would be selected if a candidate misinterprets the Control Room IA indications for which is upstream or downstream of the dryer.

d. Correct – Per ON-118-001, LOSS OF INSTRUMENT AIR, “IF Instrument Air Compressor 1K107A OR 1K107B will NOT start, OR can NOT maintain adequate pressure, THEN Ensure Unit 1 Instrument Air is cross tied to Unit 1 Service Air. NOTE: PCV 12560 should open at approximately 95 psig. Pressure downstream of the Dryers (PI 12564) will indicate approximately 85 psig.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-118-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:
Bank #
Modified Bank # LOC26R RO #20 **Parent attached**
New

Question History: Last NRC Exam January 2015

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

10 CFR Part 55 Content:
55.41 4
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
262002 UPS (AC/DC)	Tier #	2	
2.4.8 - Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	Group #	1	
	K/A #	262002	
	Importance Rating	3.8	

Proposed Question: **# 21**

Unit 1 is operating at 100% power with the following:

- Instrument Bus 1Y226 de-energizes due to a sustained electrical fault.
- ON-YPNL-101, Loss of Instrument Bus, is being performed.
- Then, a spurious Reactor scram occurs.
- The Unit Supervisor enters EO-000-102, RPV Control.

Which one of the following describes the correct procedure implementation for ON-YPNL-001?

A. Continue performing ON-YPNL-101. In the event of a conflict between ON-YPNL-101 and EO-000-102, EO-000-102 is the overriding procedure.

B. Continue performing ON-YPNL-101. In the event of a conflict between ON-YPNL-101 and EO-000-102, ON-YNPL-001 is the overriding procedure.

C. Exit ON-YPNL-101. ON-YPNL-101 is re-entered at the step in progress after exiting EO-000-102.

D. Exit ON-YPNL-101. ON-YPNL-101 entry conditions are re-evaluated after exiting EO-000-102.

Proposed Answer:	A

Explanation:

a. Correct – There is no requirement to exit ONs when EOs are entered. In fact, both procedures are executed concurrently. The EOs are higher-tiered documents than ONs, therefore in the event of a conflict, the EO must be followed.

b. Incorrect – The EOs are higher-tiered documents than ONs, therefore in the event of a conflict, the EO must be followed.

c. Incorrect – There is no requirement to exit ONs when EOs are entered. The EO is the higher-tiered document, however the ON is still performed.

d. Incorrect – There is no requirement to exit ONs when EOs are entered. The EO is the higher-tiered document, however the ON is still performed.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-AD-055 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # NMP1 2015 NRC #88 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor	Tier #	2	
2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	Group #	1	
	K/A #	215004	
	Importance Rating	4.6	

Proposed Question: **# 22**

Given the following operator actions from ON-SCRAM-101, Reactor Scram:

(1) **Ensure** Mode Switch in Shutdown.

(2) **Insert** SRMs and IRMs.

(3) **Stop** Condensate Pumps 1P102A(B)(C)(D) as necessary to leave 2 pumps in operation.

(4) **Trip** Turbine when < 150 MWe.

Which one of the following identifies which of these steps are Immediate Operator Actions for a Reactor scram, in accordance with ON-SCRAM-101?

A. (1) only.

B. (1) and (2) only.

C. (1), (2), and (3) only.

D. (1), (2), (3), and (4).

Proposed Answer:	**A**

Explanation:

a. Correct – placing the Mode Switch in Shutdown is the only Immediate Operator Action.

b. Incorrect –Inserting SRMs and IRMs is not an Immediate Operator Action.

c. Incorrect – Stopping Condensate pumps is one of the first Subsequent Operator Actions in ON-SCRAM-101, but is not an Immediate Operator Action.

d. Incorrect – Stopping Condensate pumps and tripping the Turbine are two of the first Subsequent Operator Actions in ON-SCRAM-101, but are not Immediate Operator Actions.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-SCRAM-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

400000 Component Cooling Water

A4.01 - Ability to manually operate and / or monitor in the control room: CCW indications and control

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	400000	
Importance Rating	3.1	

Proposed Question: **# 23**

Unit 1 is operating at 100% power with the following:

- TBCCW pump A and heat exchanger A are in service.
- Annunciator AR-123-G05, TBCCW HEADER HI/LO TEMP, alarms.
- TBCCW Cooler Temp TIC-10946 output indicates a downscale failure.
- TIC-10946 will not respond in manual.

Which one of the following describes:

(1) the actual TBCCW temperature, and

(2) the initial action required, in accordance with ON-TBCCW-101, Loss of Turbine Building Closed Loop Cooling Water?

A. (1) Low
(2) Throttle TBCCW Hx SW Temp CV BPV 109083 CLOSED

B. (1) Low
(2) Transfer in-service TBCCW Heat Exchanger to ESW Supply

C. (1) High
(2) Place alternate TBCCW Heat Exchanger 1E123B in service

D. (1) High
(2) Throttle TBCCW Hx SW Temp CV BPV 109083 OPEN

Proposed Answer: **D**

Explanation:

a. Incorrect – The alarm would be caused by high temperature and the ON directs opening that valve to lower temperature or return it to the normal band.

b. Incorrect – This is not the initial action, but a possible follow up action if other attempts fail.

c. Incorrect – The temperature control valve is common to both heat exchangers, so just placing an alternate heat exchanger in service will not correct the problem.

d. Correct - With indicated temperature failed low, the HX bypass valve (109083) will go closed, bypassing flow around the heat exchanger and resulting in rising TBCCW temperature. The ON directs opening that valve to lower temperature or return it to the normal band.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-TBCCW-101, TM-OP-15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #26

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis x

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
264000 EDGs	Tier #	2	
A3.01 - Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Automatic starting of compressor and emergency generator	Group #	1	
	K/A #	264000	
	Importance Rating	3.0	

Proposed Question: **# 24**

Unit 1 is shutdown with the following conditions:

- ESS Bus 1A203 is energized from its normal power source.
- Diesel Generator (DG) C is in standby.
- Then, the Unit 1 PCO manually initiates Division II of RHR.

Which one of the following describes the response of DG C?

DG C...

A. remains in standby.

B. starts and runs unloaded.

C. starts and runs loaded on ESS Bus 1A203 in parallel with its normal power source.

D. starts and runs loaded on ESS Bus 1A203 after its normal power source separates.

Proposed Answer:	B

Explanation:

a. Incorrect – Even though ESS Bus 1A203 is powered from its normal power source, initiation of Division II of RHR automatically starts DG C.

b. Correct - Initiation of Division II of RHR automatically starts DG C. Since ESS Bus 1A203 is energized from its normal power source, DG C does NOT load onto the bus, and therefore runs unloaded.

c. Incorrect - Since ESS Bus 1A203 is energized from its normal power source, DG C does NOT load onto the bus, and therefore runs unloaded.

d. Incorrect - Since ESS Bus 1A203 is energized from its normal power source, DG C does NOT load onto the bus, and therefore runs unloaded.

Technical Reference(s): OP-024-001 Section 2.8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # Vision SYSID 5428

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

262002 UPS (AC/DC)

K1.13 - Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Recirculation pump speed control: Plant-Specific

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	262002	
Importance Rating	2.5	

Proposed Question: **# 25**

Unit 1 is operating at 100% power.

Which one of the following identifies:

(1) a power source, that if lost, would affect Reactor Recirculation pump speed control, and

(2) the effect on Reactor Recirculation pump speed upon loss of this power source?

A. (1) 1Y128
(2) Pump speeds remain constant due to scoop tube positioner lock

B. (1) 1Y128
(2) Pump speeds lower due to runback signal

C. (1) 1Y236
(2) Pump speeds remain constant due to scoop tube positioner lock

D. (1) 1Y236
(2) Pump speeds lower due to runback signal

Proposed Answer:	A

Explanation:

a. Correct – Loss of 1Y128 causes both a Reactor Recirculation scoop tube positioner lock and a runback signal for both pumps. The scoop tube positioner lock keeps pump speed constant despite presence of the runback signal.

b. Incorrect – Loss of 1Y128 does cause a runback signal, however pump speed remains constant due to a simultaneous scoop tube positioner lock.

c. Incorrect – Loss of 1Y236 affects cooling to the Recirculation pump motors coolers, but not pump speed control.

d. Incorrect – Loss of 1Y236 affects cooling to the Recirculation pump motors coolers, but not pump speed control.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-117-001 ON-YPNL-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control	Tier #	2	
K4.14 - Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Selection of various instruments to provide reactor water level input	Group #	1	
	K/A #	259002	
	Importance Rating	3.4	

Proposed Question: **# 26**

Unit 2 is operating at 100% power with the following Reactor water level inputs to ICS:

- NRLA +15 inches
- NRLBB +35 inches
- NRLC +28 inches
- UPSETLB +33 inches

Which one of the following describes the ICS response to these inputs?

A. The NRLA input to the average level calculation is replaced by NRLC.

B. The NRLA input to the average level calculation is replaced by UPSETLB.

C. Level control will switch from using the average level calculation to using NRLC.

D. Level control will switch from using the average level calculation to using NRLBB.

Proposed Answer:	**A**

Explanation:

a. Correct - ICS has a level validation scheme. A low median signal is selected from NRLA, NRLBB, NRLC and UPSETLB. ICS disregards both the high signal and the low signal. Of the two median (middle) signals ICS chooses the low median for level instrument validation. In this case, the low median signal is 28", based on NRLC. If an input deviates by more than 10" from the low median signal, it is marked as DEVIANT. In this case, NRLA is more than 10" below the low median signal, and therefore DEVIANT. This causes NRLA to be replaced in the average level calculation by the low median signal of 28" from NRLC.

b. Incorrect – NRLA is replaced by the low median signal, which is 28" from NRLC.

c. Incorrect – Since only one signal (NRLA) is DEVIANT, ICS will continue to use average level calculation, just with the NRLA signal replaced. Average level calculation is only abandoned if a second NRL instrument is lost.

d. Incorrect – Since only one signal (NRLA) is DEVIANT, ICS will continue to use average level calculation, just with the NRLA signal replaced. Average level calculation is only abandoned if a second NRL instrument is lost.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-031F (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ____

Modified Bank # LOC25 Cert #25 (Note changes or attach parent)

New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
290002 Reactor Vessel Internals	Tier #	2	
K1.04 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR VESSEL INTERNALS and the following: HPCI: Plant-Specific	Group #	2	
	K/A #	290002	
	Importance Rating	3.4	

Proposed Question: **# 27**

Unit 1 has experienced a failure to scram with the following:

- Initial Reactor power after the failure to scram was 17%.
- Reactor water level has been intentionally lowered and is now being controlled in a band of -161" and -60".

Which one of the following describes where HPCI injects into the Reactor vessel and if HPCI is allowed to be used for injection in this situation, in accordance with EO-000-113, Level/Power Control?

HPCI injects into the Reactor vessel through...

A. a dedicated HPCI sparger and is allowed to be used for injection.

B. a dedicated HPCI sparger and is NOT allowed to be used for injection.

C. one of the Feedwater lines and is allowed to be used for injection.

D. one of the Feedwater lines and is NOT allowed to be used for injection.

Proposed Answer:	**C**

Explanation:

a. Incorrect - HPCI does not have a dedicated sparger. HPCI injects through Feedwater line B.

b. Incorrect - HPCI does not have a dedicated sparger. HPCI injects through Feedwater line B. Since HPCI injects outside the core shroud, it is allowed to be used for Reactor injection.

c. Correct - HPCI injects through Feedwater line B, which injects in the downcomer region outside of the core shroud. Based on this injection location, HPCI is a Preferred ATWS Injection System allowed to be used for Reactor injection in this situation.

d. Incorrect – Since HPCI injects outside the core shroud, it is allowed to be used for Reactor injection.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-52, EO-000-113 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # JAF 4/14 NRC #24

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam JAF 4/14 NRC #24

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
233000 Fuel Pool Cooling/Cleanup	Tier #	2	
K2.02 - Knowledge of electrical power supplies to the following: RHR pumps	Group #	2	
	K/A #	233000	
	Importance Rating	2.8	

Proposed Question: **# 28**

Which one of the following identifies the power supplies to the Unit 1 RHR pumps that are preferred for RHR Fuel Pool Cooling Mode, in accordance with OP-149-003, RHR Operation in Fuel Pool Cooling Mode?

A. ESS Bus 1A (1A201) and ESS Bus 1B (1A202)

B. ESS Bus 1A (1A201) and ESS Bus 1C (1A203)

C. ESS Bus 1C (1A203) and ESS Bus 1D (1A204)

D. ESS Bus 1B (1A202) and ESS Bus 1D (1A204)

Proposed Answer:	**B**

Explanation:

a. Incorrect – ESS Bus 1B (1A202) is the power supply for RHR pump 1B, which is NOT a preferred RHR pump for RHR Fuel Pool Cooling Mode.

b. Correct – RHR loop 1A is the preferred loop for RHR Fuel Pool Cooling Mode. Per OP-149-003, "This is due to capability of RHR Loop A discharge to be aligned directly to Fuel Pool Cooling and Cleanup System. If RHR Loop B is used HV 151F010A, RHR Loop A Cross Tie and HV 151F010B RHR Loop B Cross Tie must be opened and discharge routed through RHR Loop A piping, rendering A RHR Loop inoperable."

   RHR loop 1A contains RHR pumps 1A and 1C, which are powered from ESS Bus 1A (1A201) and ESS Bus 1C (1A203), respectively.

c. Incorrect – ESS Bus 1D (1A204) is the power supply for RHR pump 1D, which is NOT a preferred RHR pump for RHR Fuel Pool Cooling Mode.

d. Incorrect – ESS Bus 1B (1A202) is the power supply for RHR pump 1B, which is NOT a preferred RHR pump for RHR Fuel Pool Cooling Mode. ESS Bus 1D (1A204) is the power supply for RHR pump 1D, which is NOT a preferred RHR pump for RHR Fuel Pool Cooling Mode.

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-149-003, TM-OP-49 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
201003 Control Rod and Drive Mechanism	Tier #	2	
K3.03 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Shutdown margin	Group #	2	
	K/A #	201003	
	Importance Rating	3.2	

Proposed Question: **# 29**

Unit 1 is operating at 90% power with the following:

- A control rod pattern adjustment is in progress.
- Control rod 22-23 is given a signal to notch insert from position 24 to 22.
- When control rod 22-23 attempts to settle, the collet fingers fail to re-engage with the index tube.

Which one of the following describes the effect of the collet finger failures on shutdown margin if a Reactor scram occurs?

Given a Reactor scram prior to reset, shutdown margin...

A. is lowered, but will still meet the requirements of Technical Specifications.

B. is lowered and will NOT meet the requirements of Technical Specifications.

C. will be unaffected because control rod 22-23 fully inserts when the collet fingers fail and will remain fully inserted on a scram.

D. will be unaffected because, even though control rod 22-23 fully withdraws when the collet fingers fail, it will still fully insert on a scram.

Proposed Answer:	**D**

Explanation:

a. Incorrect – Even though control rod 22-23 initially will drift to the full out position, it will still fully insert on a Reactor scram. Therefore, the collet finger failures will NOT reduce post-scram shutdown margin.

b. Incorrect – Even though control rod 22-23 initially will drift to the full out position, it will still fully insert on a Reactor scram. Therefore, the collet finger failures will NOT reduce post-scram shutdown margin.

c. Incorrect – Even though control rod 22-23 was initially driving in, failure of the collet fingers to re-engage will result in the control rod drifting out of the core due to the pull of gravity, not into the core. Control rod 22-23 will insert once hydraulic pressure is applied to the piston during a Reactor scram.

d. Correct – During the settle portion of the control rod notch insertion sequence, the control rod should move slightly downward until the collet fingers engage the index tube and hold the control rod in place. Since the collet fingers fail to engage the index tube, the control rod drifts down and out of the core due to the pull of gravity. This will add positive reactivity to the core, however a Reactor scram will apply hydraulic pressure to the CRDM and successfully insert the control rod fully into the core. This will result in the same post-scram shutdown margin as if the collet finger failure had never occurred.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

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

Proposed references to be provided to applicants during examination:

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 6

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

215001 Traversing In-core Probe

K4.01 - Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark-I&II(Not-BWR1)

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	215001	
Importance Rating	3.4	

Proposed Question: **# 30**

Unit 1 is operating at 80% power with the following:

- Traversing In-core Probe (TIP) traces are in progress.
- The B TIP detector is stuck in the indexer.
- Then, a Main Turbine Trip occurs.
- RCIC and HPCI automatically initiate.

Which one of the following describes the status of the B TIP Ball and Shear valves five (5) minutes later?

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

Proposed Answer:	A

Explanation:

a. Correct – Automatic initiation of RCIC and HPCI indicates that Reactor water level lowered to at least -38" during the post-scram transient. PCIS automatically attempts to isolate TIPs on Reactor water level <+13". Since the B TIP detector is stuck in the indexer, the Ball valve will not close. The Shear valve is designed for just this situation, however it only closes if given a manual signal, which is not indicated in the stem conditions. Therefore, both the Shear and Ball valves remain open.

b. Incorrect – The Shear valve remains open because it does not have an automatic closure feature.

c. Incorrect – The Ball valve would be closed if the B TIP detector was not stuck. However, since the detector is stuck beyond the shield chamber, the Ball valve is still open.

d. Incorrect – The Ball valve would be closed if the B TIP detector was not stuck. However, since the detector is stuck beyond the shield chamber, the Ball valve is still open. The Shear valve remains open because it does not have an automatic closure feature.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-78F (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #33

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 9

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

286000 Fire Protection

K5.04 - Knowledge of the operational implications of the following concepts as they apply to FIRE PROTECTION SYSTEM: Valve operation

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	286000	
Importance Rating	2.9	

Proposed Question: **# 31**

Unit 1 is operating at 100% power with the following:

- A fire has been confirmed in the Lower Relay Room.
- Activation of CO_2 suppression has been directed for the Lower Relay Room.
- Power has been lost to the CO_2 suppression system due to a fault on panel 1Y629.

Which one of the following describes the required action(s) to activate CO_2 to the Lower Relay Room in accordance with OP-013-001, Fire Protection System?

A. Depress the pushbutton at the associated Push Button Station.

B. Manually open the pilot valve for the Master Select Valve, only.

C. Manually open the pilot valve for the associated Hazard Select Valve, only.

D. Manually open the pilot valves for both the Master Select Valve and the associated Hazard Select Valve.

Proposed Answer:	**C**

Explanation:

a. Incorrect – This is the normal method to manually initiate CO_2 to the Lower Relay Room, however it requires power from 1Y629 to re-position the Hazard Select Valve.

b. Incorrect – Loss of power from 1Y629 causes the normally energized pilot valve for the Master Select Valve to de-energize. This causes the Master Select Valve to open, which is the required position for activating CO_2 to the Lower Relay Room. Therefore, no action is required manually open the pilot valve for the Master Select Valve.

c. Correct – To activate CO_2 to the Lower Relay Room, both the Master Select Valve and Hazard Select Valve must be open. Upon loss of power from 1Y629, the normally energized pilot valve for the Master Select Valve de-energizes. This causes the Master Select Valve to open. The pilot valve for the Hazard Select Valve is normally de-energized and is required to energize to open the Hazard Select Valve. Since power is unavailable, the pilot valve for the Hazard Select Valve must be manually opened to activate CO_2.

d. Incorrect – Loss of power from 1Y629 causes the normally energized pilot valve for the Master Select Valve to de-energize. This causes the Master Select Valve to open, which is the required position for activating CO_2 to the Lower Relay Room. Therefore, no action is required manually open the pilot valve for the Master Select Valve.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-013-001 Section 2.8.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

215002 RBM

K6.05 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM : LPRM detectors: BWR-3,4,5

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	215002	
Importance Rating	2.8	

Proposed Question: **# 32**

Unit 1 is operating at 34% power with the following:

- Control rod 30-31 is selected.
- The LPRMs assigned to this control rod indicate as follows:

RBM A		RBM B	
LPRM	Indication	LPRM	Indication
24-25B	Bypassed	24-33B	2 %
32-33B	5 %	32-25B	2 %
24-33C	Bypassed	24-33C	Bypassed
32-25C	25 %	32-25C	25 %
24-25C	Bypassed	24-25C	Bypassed
32-33C	25 %	32-33C	25 %
24-33D	6 %	24-25D	4 %
32-25D	4 %	32-33D	2 %

Which one of the following identifies the status of the Rod Block Monitors (RBMs)?

A. Both RBMs A and B are operable.

B. Both RBMS A and B are inoperable.

C. RBM A is operable and RBM B is inoperable.

D. RBM A is inoperable and RBM B is operable.

Proposed Answer:	**C**

Explanation:

a. Incorrect – RBM B is inoperable because <50% of its assigned LPRMs are un-bypassed and >3 W/cm^2.

b. Incorrect – RBM A is operable because ≥50% of its assigned LPRMs are un-bypassed and >3 W/cm^2.

c. Correct – Each RBM must have ≥50% of its assigned LPRMs operable in order to itself be operable. An LPRM is considered operable for the RBM if it is un-bypassed and indicating >3 W/cm^2. RBM A has 5 of 8 LPRMs operable, therefore RBM A is operable. RBM B has 3 of 8 LPRMs operable, therefore RBM B is inoperable.

d. Incorrect – RBM A is operable because ≥50% of its assigned LPRMs are un-bypassed and >3 W/cm^2. RBM B is inoperable because <50% of its assigned LPRMs are un-bypassed and >3 W/cm^2.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-078K (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # Vision SYSID 34004

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
201002 RMCS	Tier #	2	
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: CRD drive water flow	Group #	2	
	K/A #	201002	
	Importance Rating	2.8	

Proposed Question: **# 33**

A Unit 1 startup is in progress with the following:

- Control rod 22-07 is given a notch withdrawal signal.
- Initially, the control rod inserts and CRD drive water flow indicates 4 gpm.
- Next, the control rod withdraws and CRD drive water flow indicates 2 gpm.
- Then, the control rod settles and CRD drive water flow indicates 0 gpm.

Which one of the following describes these CRD drive water flow indications?

A. All of these flow indications are as expected for a notch withdrawal signal.

B. While the control rod was inserting, the flow indication was higher than expected.

C. While the control rod was withdrawing, the flow indication was lower than expected.

D. While the control rod was settling, the flow indication was lower than expected.

Proposed Answer:	A

Explanation:

a. Correct – During control rod insertion, CRD drive water flow is expected to indicate 4 gpm. During control rod withdrawal, CRD drive water flow is expected to indicate 2 gpm. During control rod settling, CRD drive water flow is expected to indicate 0 gpm.

b. Incorrect – During control rod insertion, CRD drive water flow is expected to indicate 4 gpm.

c. Incorrect – During control rod withdrawal, CRD drive water flow is expected to indicate 2 gpm.

d. Incorrect – During control rod settling, CRD drive water flow is expected to indicate 0 gpm.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-156-001, TM-OP-055 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
268000 Radwaste	Tier #	2	
A2.01 - Ability to (a) predict the impacts of the following on the RADWASTE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System rupture	Group #	2	
	K/A #	268000	
	Importance Rating	2.9	

Proposed Question: **# 34**

Unit 1 is operating at 100% power with the following:

- Radwaste tanks are in a normal alignment.
- An unidentified leak has developed in the Drywell.
- Input to the Drywell Floor Drain sumps has risen from 0.3 gpm to 3.8 gpm in the last hour.
- No change has been made to Radwaste tank alignment based on this leak.

Which one of the following describes:

(1) the Radwaste tanks that will be receiving this excess leakage, and

(2) whether the leakage exceeds any Technical Specification limit?

	(1) Radwaste Tanks That Receive This Excess Leakage	(2) Leakage Exceeds Any Technical Specification Limit?
A.	Collection Tanks	No
B.	Collection Tanks	Yes
C.	Surge Tanks	No
D.	Surge Tanks	Yes

Proposed Answer:	B

Explanation:

a. Incorrect – This leakage represents a rise in Unidentified Leakage of 2.5 gpm in one hour. This exceeds the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1. The TS 3.4.4 limit of ≤ 5 gpm Unidentified Leakage is NOT exceeded.

b. Correct - The Drywell Floor Drain sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks. This leakage represents a rise in Unidentified Leakage of 2.5 gpm in one hour. This exceeds the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1.

c. Incorrect – The Drywell Floor Drain Sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks, not the Surge Tanks. This leakage represents a rise in Unidentified Leakage of 2.5 gpm in one hour. This exceeds the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1. The TS 3.4.4 limit of ≤ 5 gpm Unidentified Leakage is NOT exceeded.

d. Incorrect – The Drywell Floor Drain Sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks, not the Surge Tanks.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-069, TS 3.4.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 4

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
290001 Secondary CTMT	Tier #	2	
A3.02 - Ability to monitor automatic operations of the SECONDARY CONTAINMENT including: Normal building differential pressure: Plant-Specific	Group #	2	
	K/A #	290001	
	Importance Rating	3.5	

Proposed Question: **# 35**

Unit 1 is operating at 100% power with the following:

- Reactor Building HVAC is operating in a normal lineup.
- A substantial steam leak develops and begins raising pressure inside Zone I.

Which one of the following describes the operation of the Zone I D/P controller?

The Zone I D/P controller throttles...

A. supply flow lower.

B. supply flow higher.

C. exhaust flow lower.

D. exhaust flow higher.

Proposed Answer:	D

Explanation:

a. Incorrect – Zone I pressure is controlled by throttling exhaust flow, not supply flow.

b. Incorrect – Zone I pressure is controlled by throttling exhaust flow, not supply flow.

c. Incorrect – Exhaust flow is throttled, however higher flow is required from the exhaust to lower Zone I D/P.

d. Correct – The Zone I D/P controller throttles a damper in the RB HVAC exhaust system to control D/P. In order to lower pressure inside Zone I, the controller raises exhaust flow to a higher value.

Technical Reference(s): TM-OP-034 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP1 2009 Audit #38

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 9

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
245000 Main Turbine Gen. / Aux	Tier #	2	
A4.10 - Ability to manually operate and/or monitor in the control room: Hydrogen gas pressure	Group #	2	
	K/A #	245000	
	Importance Rating	2.6	

Proposed Question: **# 36**

A Unit 1 startup is in progress with the following:

- Main Generator load is 600 MWe, steady.
- Main Generator reactive load is 300 MVAR, steady.
- Main Generator hydrogen pressure is 38 psig, up slow.
- It is desired to raise Main Generator load to 1250 MWe while maintaining the current reactive load.

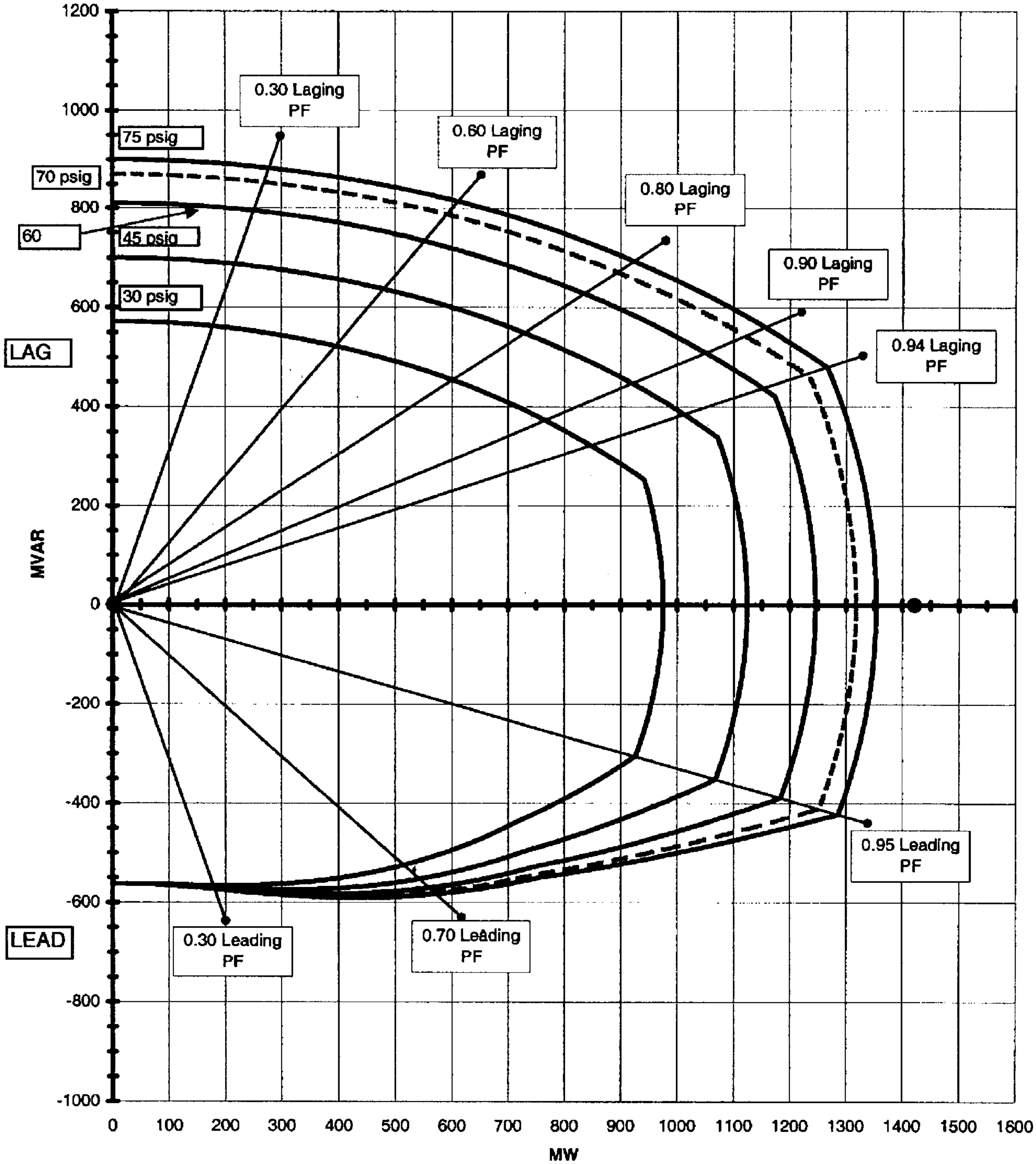
Note: The Main Generator Capability Curve is provided on the following pages.

Which one of the following identifies:

(1) the approximate highest Main Generator load allowed with the current reactive load and hydrogen pressure, and

(2) the minimum Main Generator hydrogen pressure required to allow the desired Main Generator load, in accordance with OP-198-001, Main Generator System?

	(1)	(2)
A.	875 MWe	60 psig
B.	875 MWe	70 psig
C.	1075 MWe	60 psig
D.	1075 MWe	70 psig


0.30 Laging PF
0.60 Laging PF
0.80 Laging PF
0.90 Laging PF
0.94 Laging PF
75 psig
70 psig
60
45 psig
30 psig
LAG
LEAD
MVAR
0.95 Leading PF
0.30 Leading PF
0.70 Leading PF
MW
1200
1000
800
600
400
200
0
-200
-400
-600
-800
-1000
0 100 200 300 400 500 600 700 800 900 1000 1100 1200 1300 1400 1500 1600

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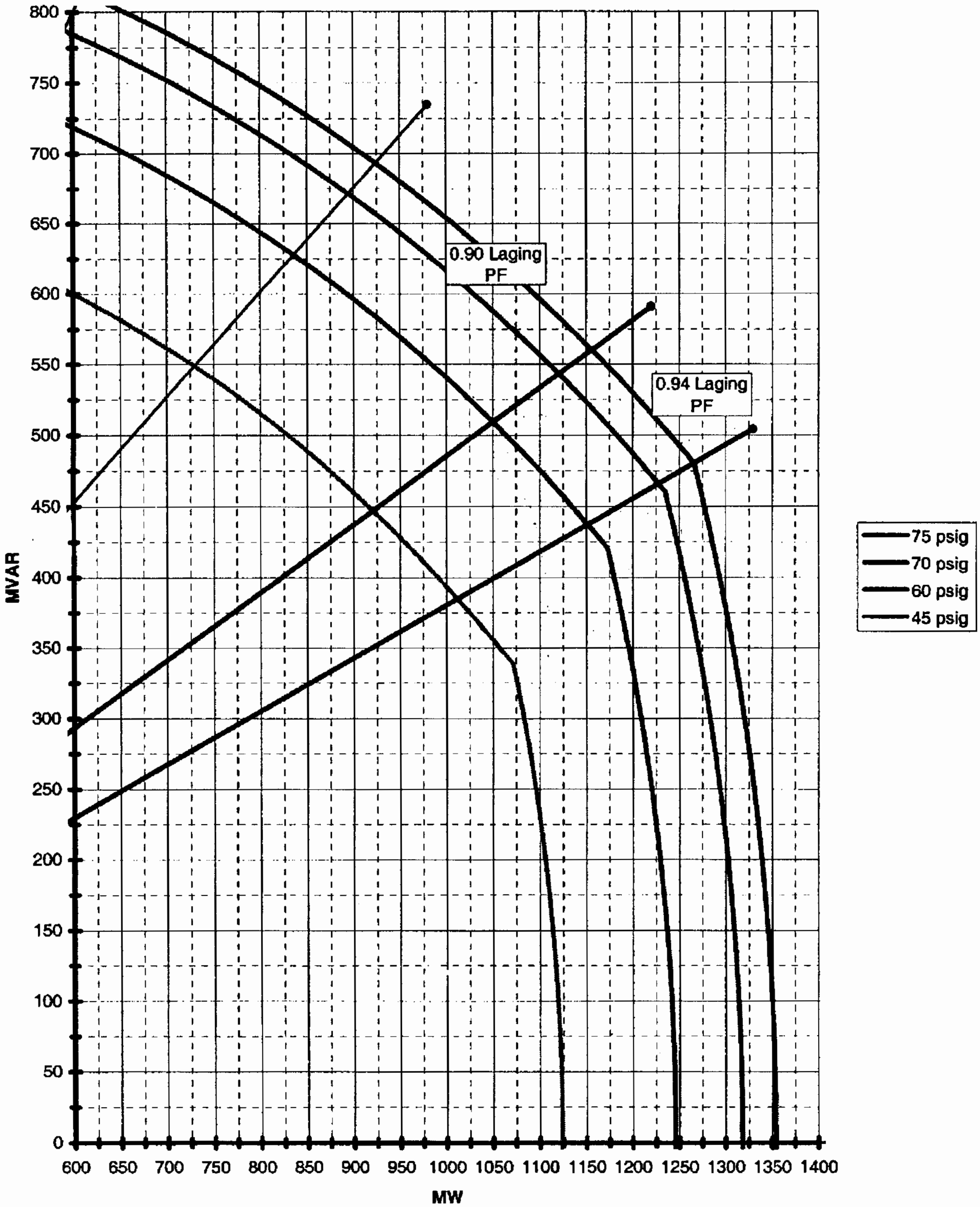

Proposed Answer:	**B**

Explanation:

a. Incorrect – To allow a combination of 1250 MWe and 300 MVAR, Main Generator hydrogen pressure must be at least 70 psig.

b. Correct - With hydrogen pressure at 38 psig, the 30 psig curve must be used. At 300 MVAR, the 30 psig curve limits Main Generator load to approximately 875 MWe. To allow a combination of 1250 MWe and 300 MVAR, Main Generator hydrogen pressure must be at least 70 psig.

c. Incorrect – With hydrogen pressure at 38 psig, the 30 psig curve must be used. At 300 MVAR, the 30 psig curve limits Main Generator load to approximately 875 MWe. To allow a combination of 1250 MWe and 300 MVAR, Main Generator hydrogen pressure must be at least 70 psig.

d. Incorrect – With hydrogen pressure at 38 psig, the 30 psig curve must be used. At 300 MVAR, the 30 psig curve limits Main Generator load to approximately 875 MWe.

Technical Reference(s): OP-198-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 5

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Comments:

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LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

201001 CRD Hydraulic

2.2.38 - Equipment Control: Knowledge of conditions and limitations in the facility license

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	201001	
Importance Rating	3.6	

Proposed Question: **# 37**

A Unit 1 startup is in progress with the following:

- Reactor pressure is 800 psig, steady.
- CRD pump 1B is out of service for maintenance.
- The following timeline of events occurs:

Time (minutes)	Event
0	CRD pump 1A trips on overcurrent. ON-CRD-101, Control Rod Malfunction, is entered. Electrical Maintenance reports that the CRD pump 1A motor has failed.
10	The first accumulator trouble alarm is received on control rod 22-23. An NPO reports control rod 22-23 accumulator pressure is 900 psig, down slow. Control rod 22-23 is at position 24.
20	The second accumulator trouble alarm is received on control rod 42-15. An NPO reports control rod 42-15 accumulator pressure is 900 psig, down slow. Control rod 42-15 is at position 48.

Which one of the following identifies the earliest time that a Reactor scram becomes required, in accordance with ON-CRD-101 and Technical Specifications?

A. Time = 10 minutes

B. Time = 20 minutes

C. Time = 30 minutes

D. Time = 40 minutes

Proposed Answer:	A

Explanation:

a. Correct – At time 10 minutes, control rod 22-23 accumulator is inoperable because its pressure is <940 psig. With Reactor pressure <900 psig and an inoperable control rod accumulator for a control rod that is not fully inserted, ON-CRD-101 and TS 3.1.5 require a manual Reactor scram immediately.

b. Incorrect – This is the time the 2nd control rod accumulator is inoperable. If Reactor pressure were >900 psig, this would start a clock for a required Reactor scram. However, with Reactor pressure <900 psig, a Reactor scram is required at the earlier time 10 minutes.

c. Incorrect – This is 20 minutes after the 1st control rod accumulator is inoperable. A 20 minute clock is involved in the scram time requirement for Reactor pressure >900 psig. However, with Reactor pressure <900 psig, a Reactor scram is required at the earlier time 10 minutes.

d. Incorrect – This is the earliest time a Reactor scram would be required with Reactor pressure >900 psig. However, with Reactor pressure <900 psig, a Reactor scram is required at the earlier time 10 minutes.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TS 3.1.5, ON-CRD-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC25 Cert #34 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
288000 Plant Ventilation	Tier #	2	
K3.05 - Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on following: Reactor building pressure: Plant-Specific	Group #	2	
	K/A #	288000	
	Importance Rating	3.1	

Proposed Question: **# 38**

Unit 1 is operating at 100% power when the running Reactor Building HVAC Zone I Exhaust fan trips due to a motor electrical fault.

Which one of the following describes the impact of this fan trip on Reactor Building HVAC?

A. Zone I D/P will stabilize near 0" WG and the running Zone I Supply fan will continue to operate.

B. Zone I D/P will stabilize near 0" WG and the running Zone I Supply fan will trip on interlock.

C. The other Zone I Exhaust fan will automatically start and recover Zone I D/P.

D. Standby Gas Treatment will automatically initiate and recover Zone I D/P.

Proposed Answer:	C

Explanation:

a. Incorrect – The standby Zone I exhaust fan automatically starts on low flow, which recovers Zone I D/P. The running Zone I Supply fan will continue to operate.

b. Incorrect – The standby Zone I exhaust fan automatically starts on low flow, which recovers Zone I D/P. The running Zone I Supply fan will continue to operate.

c. Correct – The standby Zone I exhaust fan automatically starts on low flow after the running fan trips. This automatically recovers Zone I D/P.

d. Incorrect – Standby Gas Treatment does NOT automatically start. The standby Zone I exhaust fan automatically starts on low flow, which recovers Zone I D/P. Start of Standby Gas Treatment would become necessary if the automatic start of the standby Zone I exhaust fan failed.

Technical Reference(s): TM-OP-34 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9

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Comments:

Examination Outline Cross-reference:

295006 SCRAM / 1

AK1.01 - Knowledge of the operational implications of the following concepts as they apply to SCRAM: Decay heat generation and removal

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295006	
Importance Rating	3.7	

Proposed Question: **# 39**

Unit 1 has been operating at 100% power for the past month when a Main Generator load reject occurs.

Which one of the following describes the decay heat approximately 10 seconds after the scram and the decay heat removal method?

Approximately 10 seconds after the scram, decay heat will be approximately __(1)__ of rated power. This decay heat will be removed via the __(2)__.

	(1)	(2)
A.	6%	SRVs
B.	6%	Turbine Bypass Valves
C.	9%	SRVs
D.	9%	Turbine Bypass Valves

Proposed Answer:	B

Explanation:

a. Incorrect – The Main Generator load reject causes Turbine Control Valve fast closure, but does not prevent Turbine Bypass Valve operation. The resulting transient will cause Reactor water level to lower significantly, but will not result in MSIV closure. Therefore, Turbine Bypass Valves remove decay heat, preventing Reactor pressure from reaching SRV opening setpoints.

b. Correct – The decay heat will be approximately 6% of rated power within 10 seconds of the scram. The Main Generator load reject causes Turbine Control Valve fast closure, but does not prevent Turbine Bypass Valve operation. The resulting transient will cause Reactor water level to lower significantly, but will not result in MSIV closure. Therefore, Turbine Bypass Valves remove decay heat, preventing Reactor pressure from reaching SRV opening setpoints.

c. Incorrect – The decay heat will be approximately 6% of rated power within 10 seconds of the scram. The Main Generator load reject causes Turbine Control Valve fast closure, but does not prevent Turbine Bypass Valve operation. The resulting transient will cause Reactor water level to lower significantly, but will not result in MSIV closure. Therefore, Turbine Bypass Valves remove decay heat, preventing Reactor pressure from reaching SRV opening setpoints.

d. Incorrect – The decay heat will be approximately 6% of rated power within 10 seconds of the scram.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-93L, simulator data (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 NRC #41

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC25 NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295021 Loss of Shutdown Cooling / 4	Tier #	1	
AK1.01 - Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Decay heat	Group #	1	
	K/A #	295021	
	Importance Rating	3.6	

Proposed Question: **# 40**

Unit 1 is being shutdown for a refueling outage with the following:

- The Reactor Mode Switch is in SHUTDOWN.
- The Reactor Vessel head closure bolts are still fully tensioned.
- RHR loop A is operating in Shutdown Cooling.

Then, a complete loss of Shutdown Cooling results in the following Reactor Coolant temperature response:

Time (hhmm)	Reactor Coolant Temperature (°F)
0800	113
0802	116
0804	119

Assuming the rate of temperature rise remains constant, which one of the following describes the status of the heatup rate and when a Mode change is expected?

	Status of Heatup Rate	Time of Mode Change
A.	Below Technical Specification Limit	0858
B.	Below Technical Specification Limit	0906
C.	Above Technical Specification Limit	0858
D.	Above Technical Specification Limit	0906

Proposed Answer:	A

Explanation:

a. Correct – The given data shows a heatup rate of 3°F every 2 minutes, or 90°F/hr. This is below the Technical Specification 3.4.10 heatup rate limit of 100°F/hr. The Mode change occurs when temperature reaches 200°F. At the given heatup rate, this will occur at 0858 [0804 + (200°F – 119°F) x (2 min / 3°F) = 0858].

b. Incorrect – 0906 is the time when the boiling point is reached, however the Mode change occurs earlier at 200°F.

c. Incorrect – The given data shows a heatup rate of 3°F every 2 minutes, or 90°F/hr, which is below the Technical Specification 3.4.10 heatup rate limit of 100°F/hr. A more restrictive heatup rate of 20°F/hr applies during hydrostatic testing, but not under these conditions.

d. Incorrect – 0906 is the time when the boiling point is reached, however the Mode change occurs earlier at 200°F.

Note: The question meets the K/A by testing operational implications of decay heat during a loss of Shutdown Cooling. The operational implications of decay heat are the resulting heatup rate and Mode change.

Technical Reference(s): TS Table 1.1-1, TS 3.4.10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP1 2008 NRC #54

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam NMP1 2008 NRC #54

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 5

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295004 Partial or Total Loss of DC Pwr / 6	Tier #	1	
AK1.03 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Electrical bus divisional separation	Group #	1	
	K/A #	295004	
	Importance Rating	2.9	

Proposed Question: **# 41**

Both Units are operating at 100% power when HSS-0653B, one of the four Channel 'C' Common 125V DC Load Manual Transfer Switches (for Diesel Generator ESW Valve control and indication and Diesel Generator Fuel Oil Booster Pump control), is transferred from its NORMAL position to its ALTERNATE position.

Which one of the following statements describes the impact of this transfer?

The loads powered via HSS-0653B are now powered from...

A. Unit 1 and there was NO loss of power to the affected loads.

B. Unit 2 and there was NO loss of power to the affected loads.

C. Unit 1 and there was a momentary loss of power to the affected loads.

D. Unit 2 and there was a momentary loss of power to the affected loads.

Proposed Answer:	**D**

Explanation:

a. Incorrect – Unit 1 is the normal supply, not the alternate. The switch is break-before-make to ensure proper electrical bus divisional separation. This results in a momentary loss of DC power to the loads during the transfer.

b. Incorrect – The switch is break-before-make to ensure proper electrical bus divisional separation. This results in a momentary loss of DC power to the loads during the transfer.

c. Incorrect – Unit 1 is the normal supply, not the alternate.

d. Correct – The alternate supply is from Unit 2 (2D264). The switch is break-before-make to ensure proper electrical bus divisional separation. This results in a momentary loss of DC power to the loads during the transfer.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-102-002 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC23 NRC #24

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC23 NRC #24

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1

EK2.03 - Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: ARI/RPT/ATWS: Plant-Specific

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295037	
Importance Rating	4.1	

Proposed Question: **# 42**

Unit 1 is operating at 100% power with the following:

- A manual Reactor scram has been attempted.
- RPS has failed to insert control rods.
- An Operator attempts to arm and depress both the Div 1 and Div 2 ARI manual initiation pushbuttons on 1C601.
- The Div 1 pushbutton is armed and depressed.
- The Div 2 pushbutton can NOT be armed.

Which one of the following describes the plant response?

A. All ARI vent and block valves re-position and all control rods insert.

B. Only the Div 1 ARI vent and block valves re-position and all control rods insert.

C. Only the Div 1 ARI vent and block valves re-position and control rods do NOT insert.

D. None of the ARI vent and block valves re-position and control rods do NOT insert.

Proposed Answer:	**C**

Explanation:

a. Incorrect – Only the Div 1 ARI vent and block valves re-position and control rods do NOT insert.

b. Incorrect – Control rods do NOT insert because the Div 2 ARI vent and block valves do not re-position.

c. Correct – The Div 1 pushbutton being armed and depressed causes the Div 1 ARI vent and block valves to re-position. However, both divisions of ARI vent and block valves must re-position to cause control rods to insert. Without the Div 2 pushbutton arming, the Div 2 ARI vent and block valves do not re-position. Therefore, control rods do NOT insert.

d. Incorrect – The Div 1 ARI vent and block valves re-position.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-058 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 Cert #47

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295030 Low Suppression Pool Water Level / 5	Tier #	1	
EK2.02 - Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: RCIC: Plant-Specific	Group #	1	
	K/A #	295030	
	Importance Rating	3.7	

Proposed Question: **# 43**

Unit 1 was operating at 100% power when the following occurred:

- A seismic event caused a Reactor scram.
- RCIC and HPCI are the only injection sources currently injecting into the Reactor.
- Condensate pumps are available.
- Reactor water level is -5", up slow.
- Reactor pressure is 550 psig, down slow.
- Suppression Pool water level is 16.5', down slow.

Which one of the following describes how HPCI and RCIC are to be operated, in accordance with the Emergency Operating Procedures?

A. Both RCIC and HPCI must be isolated.

B. Both RCIC and HPCI may be left running.

C. RCIC must be isolated, but HPCI may be left running.

D. RCIC may be left running, but HPCI must be isolated.

Proposed Answer: **D**

Explanation:

a. Incorrect – RCIC may be left running because it discharges a smaller amount of steam into the Suppression Pool/Chamber.

b. Incorrect – HPCI must be isolation because of the large amount of steam it discharges into the Suppression Pool/Chamber.

c. Incorrect - RCIC may be left running because it discharges a smaller amount of steam into the Suppression Pool/Chamber. HPCI must be isolation because of the large amount of steam it discharges into the Suppression Pool/Chamber.

d. Correct – EO-000-103 steps SP/L-6 and SP/L-7 require when Suppression Pool level cannot be maintained above 17', HPCI must be isolated if adequate core cooling is assured. In this case, Suppression Pool level is less than 17' and continuing to lower while Reactor water level is well above the top of active fuel with Condensate available to replace HPCI injection. Therefore, HPCI must be isolated. This is done because below 17' the HPCI exhaust discharges directly into the Suppression Chamber air space, which may cause over-pressurization. The RCIC exhaust is also discharging into the Suppression Chamber air space under these conditions, but continued use of RCIC is allowed because it discharges a much smaller amount of steam.

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LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # JAF 4/14 NRC #48

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam JAF 4/14 NRC #48

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
700000 Generator Voltage and Electric Grid Disturbances	Tier #	1	
AK2.06 - Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Reactor power	Group #	1	
	K/A #	700000	
	Importance Rating	3.9	

Proposed Question: **# 44**

Unit 1 is operating at 98% power with the following:

- Power is being raised in accordance with GO-100-002, Plant Startup, Heatup and Power Operation, and OP-164-002, Reactor Recirculation System HMI Operations.
- Reactor Recirculation pump (RRP) 1A controller is operating in Fine Speed Control mode.
- RRP 1B controller is operating in Monitor mode.
- Grid conditions have changed and the Unit 1 Main Generator is now operating 2 MWe below the capability curve.

Which one of the following describes the response of the RRP controllers?

	RRP 1A Controller	RRP 1B Controller
A.	Goes on HOLD for up to 60 minutes	Speed demand remains unchanged
B.	Terminates the power ascension and defaults to manual mode	Speed demand remains unchanged
C.	Goes on HOLD for up to 60 minutes	Lowers Reactor power in 4 MWth increments until sufficient margin has been established
D.	Terminates the power ascension and defaults to manual mode	Lowers Reactor power in 4 MWth increments until sufficient margin has been established

Proposed Answer:	**A**

Explanation:

a. Correct – When margin to the Main Generator capability curve lowers below 2.1 MWe, RRP 1A controller goes into HOLD for up to 60 minutes. This stops any immediate Reactor power rise, but will continue power ascension if adequate margin becomes available again within 60 minutes. RRP 1B controller remains in monitor mode with speed demand unchanged. RRP 1B would only lower Reactor power if the margin to the Main Generator capability curve were to further degrade to <0.1 MWe.

b. Incorrect - RRP 1A controller goes into HOLD for up to 60 minutes. The programmed power ascension on RRP 1A controller would be terminated if the margin to the Main Generator capability curve were to further degrade to <0.1 MWe.

c. Incorrect - RRP 1B controller remains in monitor mode with speed demand unchanged. RRP 1B would only lower Reactor power if the margin to the Main Generator capability curve were to further degrade to <0.1 MWe.

d. Incorrect - RRP 1A controller goes into HOLD for up to 60 minutes. The programmed power ascension on RRP 1A controller would be terminated if the margin to the Main Generator capability curve were to further degrade to <0.1 MWe. RRP 1B controller remains in monitor mode with speed demand unchanged. RRP 1B would only lower Reactor power if the margin to the Main Generator capability curve were to further degrade to <0.1 MWe.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-164-002 Section 2.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 NRC #20

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC24 NRC #20

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments: TRH 4/28/15 – Clarified that final bullet is operating below the curve, based on validator comment.

Examination Outline Cross-reference:	Level	RO	SRO
295016 Control Room Abandonment / 7	Tier #	1	
AK3.01 - Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Reactor SCRAM	Group #	1	
	K/A #	295016	
	Importance Rating	4.1	

Proposed Question: **# 45**

Which one of the following describes the reason why the Reactors are manually scrammed prior to evacuating the Main Control Room, in accordance with ON-100-109, Control Room Evacuation?

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

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

C. Scramming from outside the Control Room would require access to plant areas that may be inaccessible due to post-accident high radiation levels.

D. Scramming from outside the Control Room would require all RPS bus power to be tripped, which would also cause undesired isolations.

Proposed Answer:	B

Explanation:

a. Incorrect – The Control Room evacuation strategy is based on maintaining Reactor water level with RCIC from the RSPs, not HPCI.

b. Correct – The Reactor being manually scrammed from the Main Control Room prior to evacuation is part of the assumptions in the associated safety analysis. This condition is necessary to ensure the remote shutdown strategy of controlling Reactor level from the RSPs with RCIC is successful.

c. Incorrect – The Control Room evacuation analysis specifically does not include a simultaneous accident or natural disaster.

d. Incorrect – The Control Room evacuation analysis does provide a backup method for scramming from outside the Control Room other than tripping all RPS bus power (select circuits are tripped only). However, the analysis assumes the scram will normally be performed form the Control Room prior to evacuation.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-100-109, FSAR 7.4.1.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # Limerick 2008 NRC #6

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Limerick 2008 NRC #6

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
600000 Plant Fire On-site / 8	Tier #	1	
AK3.04 - Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site	Group #	1	
	K/A #	600000	
	Importance Rating	2.8	

Proposed Question: **# 46**

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

- A fire is detected in the Unit 2 Turbine Building.
- ON-013-001, Response to Fire, has been entered.
- Simplex alarm FIRE DET 106_Z4 ALM, Control Structure Outside Air Intake, has actuated.
- A slight smell of smoke is detected in the Main Control Room.
- The Fire Brigade reports that the fire is under control but still in progress.

Which one of the following describes the required control of the CREOASS and the associated reason, in accordance with ON-013-001?

Place CREOASS in the...

A. RECIRCULATION mode to ensure continued Main Control Room habitability.

B. RECIRCULATION mode to ensure continued Main Control Room equipment operability.

C. PRESSURIZATION/FILTRATION mode to ensure continued Main Control Room habitability.

D. PRESSURIZATION/FILTRATION mode to ensure continued Main Control Room equipment operability.

Proposed Answer:	A

Explanation:

a. Correct – With the given Simplex alarm and smoke detected in the Main Control Room, ON-013-001 requires placing CREOASS in the recirculation mode. The basis for this action is to maintain habitability for Control Room personnel.

b. Incorrect – The basis for placing CREOASS in the recirculation mode is specifically to maintain habitability for Control Room personnel, not to maintain Control Room equipment operability.

c. Incorrect – ON-013-001 requires placing CREOASS in the recirculation mode, not the pressurization/filtration mode.

d. Incorrect – ON-013-001 requires placing CREOASS in the recirculation mode, not the pressurization/filtration mode. The basis for placing CREOASS in the recirculation mode is specifically to maintain habitability for Control Room personnel, not to maintain Control Room equipment operability.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-013-001 Section 5.6 and Attachment B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 Cert #77 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-site Release Rate / 9	Tier #	1	
EK3.02 - Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System isolations	Group #	1	
	K/A #	295038	
	Importance Rating	3.9	

Proposed Question: **# 47**

Unit 1 is operating at 100% power with the following:

- Annunciator AR-106-G03, OFF-GAS HI RADIATION, alarms.
- Annunciator AR-111-C03, MN STM LINE RAD MONITOR HI RADIATION, alarms.
- Both alarms have been determined to be valid.
- NO other alarms have yet been received.
- Turbine Building Ventilation SPING indications are slightly elevated and up slow.
- ON-179-001, Increasing Offgas / MSL Rad Level, has been entered with no additional actions taken.

Which one of the following describes the need to scram the Reactor and isolate the Main Steam Lines, in accordance with ON-179-001?

A Reactor scram and isolation of the Main Steam lines is...

A. required because the Offgas hi radiation alarm has been received.

B. required because the Main Steam Line hi radiation alarm has been received.

C. NOT required because the Offgas hi-hi radiation alarm has NOT been received.

D. NOT required because the Main Steam Line hi-hi radiation alarms have NOT been received.

Proposed Answer:	**D**

Explanation:

a. Incorrect – ON-179-001 does not yet require a scram and MSL isolation.

b. Incorrect – ON-179-001 does not yet require a scram and MSL isolation.

c. Incorrect – The requirement to scram and isolate MSLs in ON-179-001 is based on MSL hi-hi radiation, not Offgas hi-hi radiation.

d. Correct – ON-179-001 provides guidance for responding to a fuel failure in order to minimize or prevent an off-site release. ON-179-001 does not yet require a scram and MSL isolation based on the conditions given. This is only required if the MSL hi-hi radiation alarm setpoint is exceeded.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-179-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10

55.43 ______

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295025 High Reactor Pressure / 3

EA1.01 - Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Main steam line drains

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295025	
Importance Rating	2.9	

Proposed Question: **# 48**

Unit 1 has experienced an accident with the following:

- Reactor water level is -170", down slow.
- Emergency RPV Depressurization is required.
- Only 1 SRV can be opened.
- Reactor pressure is 1050 psig, down slow.

The Unit Supervisor has directed the use of Main Steam Line Drains to assist in performing Emergency RPV Depressurization.

Which one of the following describes the ability to use the Main Steam Line Drains for Reactor pressure reduction under these conditions?

Main Steam Line Drains...

A. can be used, but only after bypassing interlocks.

B. can be used without the need to bypass interlocks.

C. CANNOT be used because the Main Condenser is NOT available.

D. CANNOT be used when Emergency RPV Depressurization is required.

Proposed Answer: **A**

Explanation:

a. Correct – With < 5 SRVs open and Reactor pressure still high, EO-000-112, Emergency RPV Depressurization, directs use of additional systems to lower Reactor pressure. The MSL drains are included in the authorized list in Table P-2. With Reactor water level <-129", MSIVs and MSL drains are closed due to an automatic isolation. However, EO-000-112 directs use of ES-184-002 to bypass this isolation and re-open MSL drains.

b. Incorrect – With Reactor water level <-129", the MSL drains are isolated. Therefore, interlocks must be bypassed to re-open the MSL drains.

c. Incorrect – Although the Main Condenser is currently not available with MSIVs closed due to low Reactor water level, EO-000-112 and ES-184-002 provide the appropriate direction to use MSL drains for Reactor pressure reduction.

d. Incorrect – MSL drains are not normally part of the Emergency RPV Depressurization strategy. However, EO-000-112 does specifically authorize use of MSL drains for Reactor pressure reduction in this situation since <5 SRVs are open and Reactor pressure is still high.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-112, ES-184-001, ON-159-002 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 Cert #50 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

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

Proposed Question: **# 49**

Refueling is in progress on Unit 1 with the following:

- The Unit 1 Refuel Platform is over the core with a fuel bundle suspended from the fuel grapple.
- A control rod begins drifting out.

Which one of the following identifies the icon that will be lit on the Refueling Platform Interlock Status Display?

A. Fault Lockout

B. Backup Hoist Limit

C. Safety Travel Interlock

D. Bridge Reverse Stop #1

Proposed Answer:	D

Explanation:

a. Incorrect – The Fault Lockout only lights if various motor overloads/overspeeds occur, or if operational timer checks fail.

b. Incorrect – The Backup Hoist Limit icon only lights if the upper limit switch on the fuel grapple is reached (in the event the normal-up limit switch fails).

c. Incorrect – The Safety Travel Interlock icon only lights if the Refuel Platform is outside of the normal boundary.

d. Correct – Bridge Reverse Stop #1 lights due to a control rod being withdrawn with the fuel grapple loaded over the core.

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-181-001 Attachment O (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #58

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 13

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295005 Main Turbine Generator Trip / 3	Tier #	1	
AA1.01 - Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system: Plant-Specific	Group #	1	
	K/A #	295005	
	Importance Rating	3.1	

Proposed Question: **# 50**

Unit 1 is operating at 75% power with the following:

- The Main Turbine spuriously trips.
- The Reactor scrams.
- Reactor pressure reaches a peak of 1100 psig during the transient.
- Reactor pressure is now 935 psig, steady, on Turbine Bypass Valves.
- Reactor water level reaches a low of -25" during the transient.
- Reactor water level is now 35", steady, with Feedwater controlling.

Which one of the following describes the response of the Reactor Recirculation pumps?

The Reactor Recirculation pumps...

A. trip.

B. remain running at the original speed.

C. remain running, but run back to Speed Limiter #2 (48%).

D. remain running, but run back to Speed Limiter #1 (30%).

Proposed Answer:	**A**

Explanation:

a. Correct – A Main Turbine trip from high power (>26%) actuates the EOC-RPT logic, which trips both Recirc pumps.

b. Incorrect - A Main Turbine trip from high power (>26%) actuates the EOC-RPT logic, which trips both Recirc pumps.

c. Incorrect - A Main Turbine trip from high power (>26%) actuates the EOC-RPT logic, which trips both Recirc pumps. On other scram signals, the Recirc pumps remain running and run back to 30% speed due to Reactor water level <+13".

d. Incorrect - A Main Turbine trip from high power (>26%) actuates the EOC-RPT logic, which trips both Recirc pumps. On other scram signals, the Recirc pumps remain running and run back to 30% speed due to Reactor water level <+13".

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-058 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295031 Reactor Low Water Level / 2	Tier #	1	
EA2.02 - Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor power	Group #	1	
	K/A #	295031	
	Importance Rating	4.0	

Proposed Question: **# 51**

Unit 1 is operating at 100% power with the following:

- An automatic Reactor scram occurs.
- Multiple control rods fail to insert.
- Reactor Recirculation pumps are tripped.
- Standby Liquid Control injection is started.
- Reactor power is 13%, steady.
- Reactor water level is 0", up slow.
- The level leg of EO-000-113, Level/Power Control, is being executed.

Which one of the following describes the required control of Reactor water level, in accordance with EO-000-113?

Reactor water level...

A. may be restored to a band of 13" to 54".

B. must be lowered to at least -60". Reactor water level may first be restored to a band of 13" to 54" when Reactor power drops below 5%.

C. must be lowered to at least -60". Reactor water level may first be restored to a band of 13" to 54" when Hot S/D Boron Weight has been injected.

D. must be lowered to at least -60". Reactor water level may first be restored to a band of 13" to 54" when Cold S/D Boron Weight has been injected.

Proposed Answer:	C

Explanation:

a. Incorrect – With Reactor power >5%, the level leg of EO-000-113 requires stopping injection into the Reactor and intentionally lowering Reactor water level to at least -60".

b. Incorrect – Reactor power lowering <5% factors into the decision of how far to lower Reactor water level, but dropping <5% alone does not allow restoring level to a band of 13" to 54" inches.

c. Correct – With Reactor power >5%, the level leg of EO-000-113 requires stopping injection into the Reactor and intentionally lowering Reactor water level to at least -60". Restoration of Reactor water level to a band of 13" to 54" requires injection of Hot Shutdown Boron Weight.

d. Incorrect - Restoration of Reactor water level to a band of 13" to 54" is first allowed when Hot Shutdown Boron Weight has been injected. Cold Shutdown Boron Weight is required for Reactor depressurization.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-113 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295018 Partial or Total Loss of CCW / 8	Tier #	1	
AA2.03 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Cause for partial or complete loss	Group #	1	
	K/A #	295018	
	Importance Rating	3.2	

Proposed Question: **# 52**

Unit 1 is operating at 100% power with the following:

- Annunciator AR-123-E06, RBCCW HEAD TANK HI LO LEVEL, alarms.
- An Operator reports current tank level is 7 feet.

Which one of the following is a possible cause of this problem?

A tube leak in the...

A. Reactor Building sump cooler

B. Control Rod Drive pump cooler

C. Reactor Water Cleanup regenerative heat exchanger

D. Reactor Water Cleanup non-regenerative heat exchanger

Proposed Answer:	**D**

Explanation:

a. Incorrect – RBCCW head tank level is high, not low. RBCCW system pressure is higher than the pressure of the water in the Reactor Building sumps, therefore any leakage in this cooler would result in lowering RBCCW head tank level, not high level.

b. Incorrect – CRD pump coolers are supplied cooling water from TBCCW, not RBCCW.

c. Incorrect – The RWCU RHX is cooled by RWCU flow returning to the Reactor, not RBCCW. The NRHX is cooled by RBCCW.

d. Correct – The given indications show that RBCCW head tank level is high (>6'1"). For this to be caused by a tube leak, the leak must occur in a heat exchanger where the system's pressure is greater than RBCCW pressure. The RWCU NRHX is cooled by RBCCW. RWCU system pressure is higher than RBCCW pressure, therefore a leak in the NRHX would cause RBCCW head tank level to rise.

Technical Reference(s): AR-123-E06, ON-114-001 Section 3.8 ON-RBCCW-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # NMP1 2010 NRC #55 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295019 Partial or Total Loss of Inst. Air / 8	Tier #	1	
AA2.02 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	Group #	1	
	K/A #	295019	
	Importance Rating	3.6	

Proposed Question: **# 53**

Which one of the following describes how the Feedwater system would be affected by a complete loss of Instrument Air while operating at 100% power?

The Feedwater pump minimum flow recirculation valves, FV–10604A(B)(C), would fail __(1)__ and the Feedwater low–load bypass valve, HV–10640, would fail __(2)__.

	(1)	(2)
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

Proposed Answer:	**B**

Explanation:

a. Incorrect - The low–load bypass valve fails CLOSED.

b. Correct – Feedwater minimum flow recirculation valves fail OPEN, and the low–load bypass valve fails CLOSED.

c. Incorrect - Feedwater minimum flow recirculation valves fail OPEN. The low–load bypass valve fails CLOSED.

d. Incorrect - Feedwater minimum flow recirculation valves fail OPEN.

SUSQUEHANNA STEAM ELECTRIC STATION
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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-118-001 Attachment A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 NRC #42

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC25 NRC #42

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295003 Partial or Complete Loss of AC / 6

2.4.1 - Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295003	
Importance Rating	4.6	

Proposed Question: **# 54**

Unit 1 is operating at 30% power with the following:

- A loss of all offsite power occurs.
- The Main Generator trips on reverse power.
- All Diesel Generators fail to start automatically and manually.
- All Uninterruptable Power Supplies and DC sources operate properly.
- Reactor pressure is 1100 psig, up slow.
- Reactor water level is 5", up slow.
- Drywell temperature is 135°F, up slow.
- Drywell pressure is 0.7 psig, up slow.

Given the following Emergency Operating Procedures:

(1) EO-000-102, RPV Control

(2) EO-000-103, Primary Containment Control

(3) EO-100-030, Unit 1 Response to Station Blackout

Which one of the following identifies which of these Emergency Operation Procedures must be entered?

A. (1) only

B. (3) only

C. (1) and (3) only

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

Proposed Answer:	**C**

Explanation:

a. Incorrect - EO-100-030 must also be entered due to loss of all offsite power and all four Unit 1 ESS buses being de-energized (due to failure of all four DGs). The remaining on-site DC power (and AC power from UPSs) does not prevent the need for entering EO-100-030.

b. Incorrect – EO-000-102 must also be entered due to low Reactor water level (<13") and high Reactor pressure (>1087 psig). The presence of a Station Blackout does not prevent the need for entering EO-000-102. Rather, EO-000-102 and EO-100-030 are performed concurrently.

c. Correct - EO-000-102 must be entered due to low Reactor water level (<13") and high Reactor pressure (>1087 psig). EO-100-030 must also be entered due to loss of all offsite power and all four Unit 1 ESS buses being de-energized (due to failure of all four DGs). Both Drywell temperature and pressure are elevated due to the Station Blackout, but neither has reached a level requiring EO-000-103 entry yet (150°F and 1.72 psig, respectively).

d. Incorrect - Both Drywell temperature and pressure are elevated due to the Station Blackout, but neither has reached a level requiring EO-000-103 entry yet (150°F and 1.72 psig, respectively).

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-102, EO-000-103, EO-100-030 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:

295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

2.1.32 - Conduct of Operations: Ability to explain and apply all system limits and precautions

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295001	
Importance Rating	3.8	

Proposed Question: **# 55**

Unit 1 is operating at 100% power with the following:

- Reactor Recirculation pump 1A trips.
- Reactor power lowers to 58%.
- Core flow indicates 48 Mlbm/hr.
- Reactor Recirculation pump 1B speed is 48%.

Which one of the following describes the required method of determining actual core flow, in accordance with ON-RECIRC-101, Reactor Recirculation Malfunction?

Determine actual core flow using...

A. core pressure drop because of the current Reactor power level.

B. core flow indication because of the current Reactor power level.

C. core pressure drop because of the current Recirculation pump speed.

D. core flow indication because of the current Recirculation pump speed.

Proposed Answer:	C

Explanation:

a. Incorrect – Actual core flow must be determined using core pressure drop, but it is because of the current Recirculation pump speed, not the current Reactor power.

b. Incorrect – Actual core flow must be determined using core pressure drop because Recirculation pump 1B speed is <75%.

c. Correct – ON-RECIRC-101 Note B.4 states, "When operating recirculation pump speed is < 75% rated speed (1260 rpm), Core Pressure Drop (i.e., Core Plate DP) is used to determine actual core flow due to inaccuracies associated with indicated core flow." In this case, actual core flow must be determined using core pressure drop because Recirculation pump 1B speed is <75%.

d. Incorrect - Actual core flow must be determined using core pressure drop, not core flow indication, because Recirculation pump 1B speed is <75%.

Technical Reference(s): ON-RECIRC-101 Note B.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

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

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis x

10 CFR Part 55 Content:
55.41 5
55.43 ____

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295026 Suppression Pool High Water Temp. / 5	Tier #	1	
2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes	Group #	1	
	K/A #	295026	
	Importance Rating	3.8	

Proposed Question: **# 56**

Unit 1 has experienced a transient with the following:

- Suppression Pool water level is 20', steady.
- Suppression Pool water temperature indications are:
  - Average SPOTMOS: 100°F
  - Bottom SPOTMOS: 120°F

Which one of the following identifies which SPOTMOS indication should be used to determine Suppression Pool water temperature and whether this indication is above or below the temperature requiring a Reactor scram, in accordance with EO-000-103, Primary Containment Control?

Determine Suppression Pool water temperature using ___(1)___ SPOTMOS indication.

This indication is currently ___(2)___ the temperature requiring a Reactor scram.

	(1)	(2)
A.	Average	above
B.	Average	below
C.	Bottom	above
D.	Bottom	below

Proposed Answer:	C

Explanation:

a. Incorrect – Since Suppression Pool water level is below 20.5', Suppression Pool water temperature should be determined using Bottom SPOTMOS, not Average.

b. Incorrect – Since Suppression Pool water level is below 20.5', Suppression Pool water temperature should be determined using Bottom SPOTMOS, not Average.

c. Correct – A note in EO-000-103 directs using Bottom SPOTMOS to determine Suppression Pool water temperature when Suppression Pool water level is below 20.5'. This is due to some of the sensors that input to the Average SPOTMOS calculation becoming uncovered at 20.5'. With current Suppression Pool water level at 20', the Bottom SPOTMOS indication should be used. The currently Bottom SPOTMOS indication is above the 110°F limit requiring a Reactor scram.

d. Incorrect – The currently Bottom SPOTMOS indication is above the 110°F limit requiring a Reactor scram.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s):	EO-000-103	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	x	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295028 High Drywell Temperature / 5

EK2.04 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295028	
Importance Rating	3.6	

Proposed Question: **# 57**

Unit 1 has experienced a steam leak in the Drywell with the following:

- Drywell pressure is 5 psig, up slow.
- Drywell temperature is 170°F, up slow.
- Suppression Chamber pressure is 3 psig, up slow.

Which one of the following describes the current status of Drywell coolers, and the required control of Drywell coolers, in accordance with EO-000-103, Primary Containment Control?

Drywell coolers are...

A. operating, but should be shut down.

B. shut down, but should be restarted.

C. operating and should be maintained operating.

D. shut down and should be maintained shutdown.

Proposed Answer:	**B**

Explanation:

a. Incorrect – Drywell coolers are normally operating, but have automatically shut down due to Drywell pressure >1.72 psig.

b. Correct – The Drywell coolers and fans have automatically shut down due to a LOCA signal (Drywell pressure >1.72 psig). With Drywell temperature above 150°F, EO-000-103 requires operating all available Drywell coolers, bypassing isolations as necessary. Drywell coolers are only required to be shut down when Drywell spray is initiated. Drywell spray is not required yet since Suppression Chamber pressure is below 13 psig and Drywell temperature is well below 340°F.

c. Incorrect – Drywell coolers are normally operating, but have automatically shut down due to Drywell pressure >1.72 psig.

d. Incorrect - The Drywell coolers and fans have automatically shut down due to a LOCA signal (Drywell pressure >1.72 psig). With Drywell temperature above 150°F, EO-000-103 requires operating all available Drywell coolers, bypassing isolations as necessary.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-159-002, OP-160-001 Section 2.5, EO-000-103, ES-134-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295024 High Drywell Pressure / 5	Tier #	1	
2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	Group #	1	
	K/A #	295024	
	Importance Rating	4.3	

Proposed Question: **# 58**

Unit 1 has experienced a loss of coolant accident with the following:

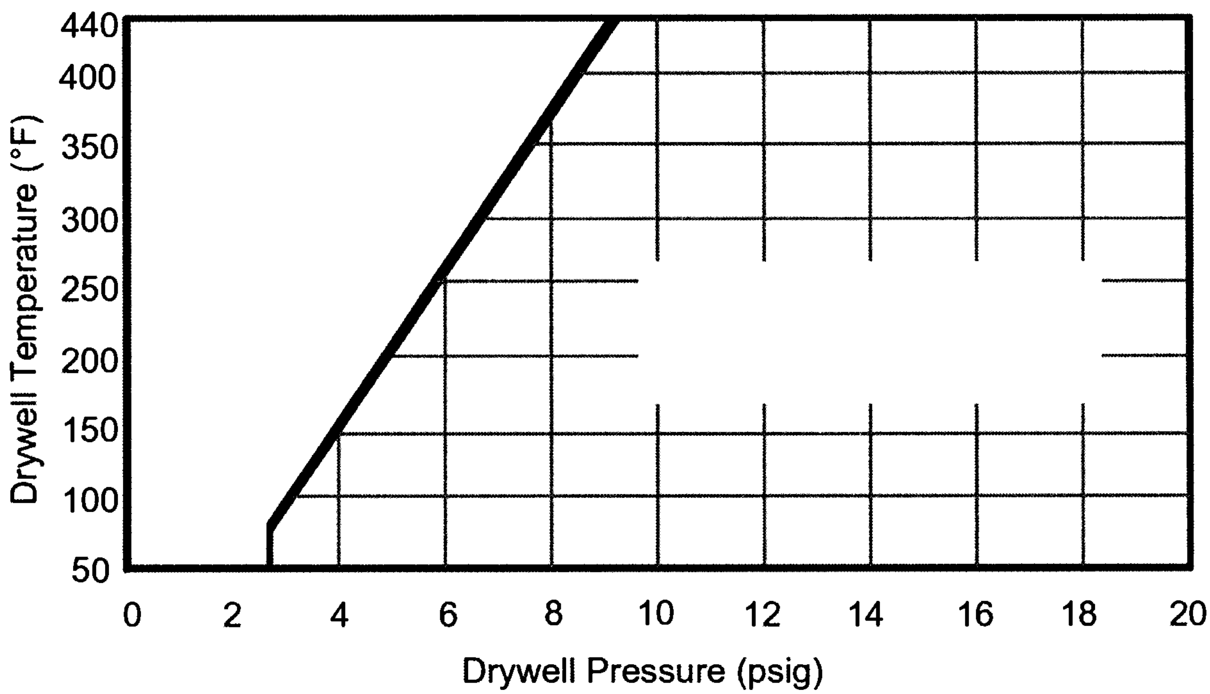
- HPCI is injecting to the Reactor.
- Reactor water level is 13", up slow.
- Reactor pressure is 800 psig, down slow.
- Drywell pressure is 19 psig, up slow.
- Drywell average temperature is 225°F, up slow.
- Suppression Chamber pressure is 14 psig, up slow.
- Suppression Pool water temperature is 100°F, up slow.

Note: A portion of EO-000-103, Primary Containment Control, is provided on the following page.

Which one of the following describes the need for Suppression Chamber and Drywell sprays, in accordance with EO-000-103, Primary Containment Control?

	Suppression Chamber Spray	Drywell Spray
A.	NOT required	NOT required
B.	NOT required	Required
C.	Required	NOT required
D.	Required	Required

FIG. 6 DSIL
DRYWELL SPRAY INITIATION LIMIT



Proposed Answer:	D

Explanation:

a. Incorrect – Both Suppression Chamber spray and Drywell spray are required under these conditions.

b. Incorrect – Since Drywell pressure is above 1.72 psig and adequate core cooling is assured, Drywell spray is required.

c. Incorrect – Since Suppression Chamber pressure is >13 psig, Figure 6 is satisfied, and adequate core cooling is assured, Drywell spray is required.

d. Correct – Suppression Chamber spray is required when Drywell pressure exceeds 1.72 psig and is not required to be stopped until before Suppression Chamber pressure drops to 0 psig. Drywell spray is required when Suppression Chamber pressure exceeds 13 psig and is not required to be stopped until before Drywell pressure drops to 0 psig. Figure 6, Drywell Spray Initiation Limit, is satisfied by current conditions, therefore it does not prevent the need for Drywell spray. Adequate core cooling is assured with HPCI injecting and Reactor water level of 13" and rising, so RHR is not otherwise needed for injection.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #52

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295017 High Off-site Release Rate / 9	Tier #	1	
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Protection of the general public	Group #	2	
	K/A #	295017	
	Importance Rating	3.8	

Proposed Question: **# 59**

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

- A Main Steam line break occurs in the Turbine Building.
- The Reactor is scrammed.
- MSIVs fail to close.
- EO-200-105, Radioactivity Release Control, is entered.
- The running Turbine Building HVAC fans trip.

Which one of the following describes how Turbine Building HVAC must be controlled and the reason why, per EO-200-105?

Turbine Building HVAC must be...

A. re-started to minimize an unmonitored ground level release.

B. left out of service to minimize the total release to the environment.

C. re-started to maintain Turbine Building equipment temperatures within limits.

D. left out of service to prevent damage to the redundant train of Turbine Building HVAC.

Proposed Answer:	A

Explanation:

a. Correct - EO-200-105 step RR-1 directs re-starting Turbine Building HVAC. This is done to minimize the unmonitored ground level release by directing steam to the stack, an elevated, monitored release. This is done to assist in dispersing any radioactivity released and delaying the radioactivity from affecting populations before protective actions can be carried out.

b. Incorrect – Turbine Building HVAC is required to be re-started.

c. Incorrect – The reason for re-starting Turbine Building HVAC it to minimize unmonitored ground level release, not to control Turbine Building temperatures below EQ program requirements.

d. Incorrect – Turbine Building HVAC is required to be re-started.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-200-105 and bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC25 Cert #59

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295034 Secondary Containment Ventilation High Radiation / 9	Tier #	1	
EK2.03 - Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following: SBGT/FRVS: Plant-Specific	Group #	2	
	K/A #	295034	
	Importance Rating	4.3	

Proposed Question: **# 60**

Unit 1 is operating at 100% power with the following:

- An irradiated fuel bundle is damaged in the Unit 1 Fuel Pool.
- Both Unit 1 Refuel Floor Wall Exhaust rad monitors indicate 25 mR/hr, up slow.
- Both Unit 1 Refuel Floor High Exhaust rad monitors indicate 30 mR/hr, up slow.

Which one of the following describes the status of the Standby Gas Treatment system?

The Standby Gas Treatment system...

A. remains in standby.

B. automatically initiates due to a Zone 1 isolation signal, only.

C. automatically initiates due to a Zone 3 isolation signal, only.

D. automatically initiates due to both Zone 1 and Zone 3 isolation signals.

Proposed Answer:	C

Explanation:

a. Incorrect – SBGT automatically initiates due to a Zone 3 isolation signal.

b. Incorrect – SBGT automatically initiates due to a Zone 3 isolation signal, not a Zone 1 isolation signal.

c. Correct – Refuel Floor Wall Exhaust radiation >21 mR/hr and Refuel Floor High Exhaust radiation >18 mR/hr both exceed the associated high alarm setpoints. Both of these conditions cause a Zone 3 isolation signal, which automatically initiates SBGT. No Zone 1 isolation signal is received since there is no indication of high Drywell pressure or low Reactor water level.

d. Incorrect – A Zone 1 isolation signal does NOT occur.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-070 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # Vision SYSID 6314 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 9

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
295009 Low Reactor Water Level / 2	Tier #	1	
AK3.02 - Knowledge of the reasons for the following responses as they apply to LOW REACTOR WATER LEVEL: Reactor feedpump runout flow control: Plant-Specific	Group #	2	
	K/A #	295009	
	Importance Rating	2.7	

Proposed Question: **# 61**

Unit 1 is operating at 80% power with the following:

- The Reactor is manually scrammed.
- Reactor water level reaches a low of -35".

Which one of the following describes an automatic Feedwater Level Control response to these conditions and the reason for this response?

A. Total Feedwater flow is limited to 3,000 gpm to prevent overfilling the Reactor.

B. Total Feedwater flow is limited to 3,000 gpm to prevent tripping pumps on low suction pressure.

C. Individual Feedwater pump flows are limited to 16.4% of rated to prevent overfilling the Reactor.

D. Individual Feedwater pump flows are limited to 16.4% of rated to prevent tripping pumps on low suction pressure.

Proposed Answer:	A

Explanation:

a. Correct – When the Reactor is scrammed and Reactor water level lowers below +13", the Setpoint Setdown feature of the Feedwater Level Control system is initiated. Part of the Setpoint Setdown logic limits total Feedwater flow rate to 3,000 gpm as part of the strategy to prevent overfilling the Reactor, which would complicate recovery and trip Feedwater pumps on high level.

b. Incorrect – The reason is to prevent Reactor overfill, not to limit Feedwater pump suction pressure. A separate RFP Suction Pressure feature exists to limit individual pump flow if low suction pressure is detected.

c. Incorrect – Setpoint Setdown limits the total Feedwater flow to <3,000 gpm, not individual flows to <16.4% of rated. Individual flows of 16.4% of rated are associated with the Recirculation pump runback logic.

d. Incorrect – Setpoint Setdown limits the total Feedwater flow to <3,000 gpm, not individual flows to <16.4% of rated. Individual flows of 16.4% of rated are associated with the Recirculation pump runback logic. The reason is to prevent Reactor overfill, not to limit Feedwater pump suction pressure. A separate RFP Suction Pressure feature exists to limit individual pump flow if low suction pressure is detected.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-045 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295015 Incomplete SCRAM / 1	Tier #	1	
AA1.01 - Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: CRD hydraulics	Group #	2	
	K/A #	295015	
	Importance Rating	3.8	

Proposed Question: **# 62**

Unit 1 has experienced a failure to scram with the following:

- The Unit Supervisor has directed an Operator to maximize CRD to drift rods using OP-155-001, Control Rod Drive Hydraulic System.
- CRD pumps 1A and 1B are running.
- CRD flow control valve (FCV) 1A is in service.
- CRD flow controller FC-C12-1R600 is in AUTOMATIC.

Which one of the following describes the required actions to maximize CRD to drift rods, in accordance with OP-155-001?

A. Adjust the FC-C12-1R600 AUTOMATIC setpoint to maximum. Fully open PV-146-F003, DRIVE WATER PRESS THTLG valve.

B. Adjust the FC-C12-1R600 AUTOMATIC setpoint to maximum. Fully close PV-146-F003, DRIVE WATER PRESS THTLG valve.

C. Place FC-C12-1R600 in MANUAL and fully open CRD FCV 1A. Fully open PV-146-F003, DRIVE WATER PRESS THTLG valve.

D. Place FC-C12-1R600 in MANUAL and fully open CRD FCV 1A. Fully close PV-146-F003, DRIVE WATER PRESS THTLG valve.

Proposed Answer:	C

Explanation:

a. Incorrect – FC-C12-1R600 is placed in MANUAL so that the high flow to the HCUs does not result in closure of the CRD FCV.

b. Incorrect – FC-C12-1R600 is placed in MANUAL so that the high flow to the HCUs does not result in closure of the CRD FCV. PV-146-F003 is fully opened, not closed, to supply maximum pressure to the CRD cooling water header.

c. Correct – Maximizing CRD to drift rods requires getting maximum pressure/flow to the CRD cooling water header. This is accomplished by placing FC-C12-1R600 in MANUAL and using it to fully open CRD FCV 1A, then fully opening the drive water pressure control valve, PV-146-F003.

d. Incorrect – PV-146-F003 is fully opened, not closed, to supply maximum pressure to the CRD cooling water header.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-113, OP-155-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 Cert #37

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Water Level / 5	Tier #	1	
EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Reactor pressure	Group #	2	
	K/A #	295029	
	Importance Rating	3.5	

Proposed Question: **# 63**

Unit 1 has experienced a failure to scram with an initial ATWS power of 4%.

Note: The Heat Capacity Temperature Limit is provided on the following page.

Which one of the following sets of conditions would require Emergency RPV Depressurization due to exceeding the Heat Capacity Temperature Limit, in accordance with EO-000-103, Primary Containment Control?

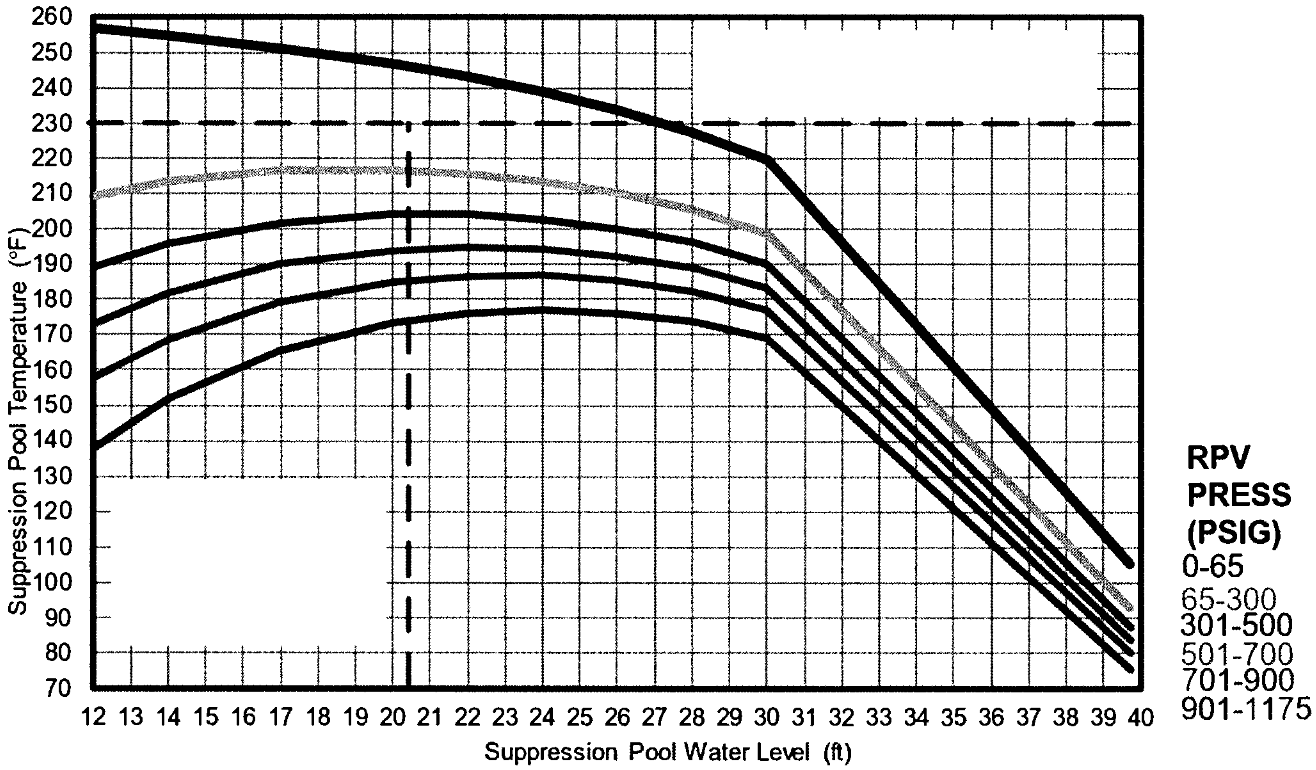
A. Reactor pressure 250 psig
Suppression Pool temperature 160°F
Suppression Pool water level 33'

B. Reactor pressure 850 psig
Suppression Pool temperature 180°F
Suppression Pool water level 27'

C. Reactor pressure 450 psig
Suppression Pool temperature 190°F
Suppression Pool water level 29'

D. Reactor pressure 750 psig
Suppression Pool temperature 170°F
Suppression Pool water level 31'

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT



Proposed Answer:	D

Explanation:

a. Incorrect – This combination of Suppression Pool temperature and water level is below the required RPV pressure line (65-300 psig – gray).

b. Incorrect – This combination of Suppression Pool temperature and water level is below the required RPV pressure line (701-900 psig – blue).

c. Incorrect – This combination of Suppression Pool temperature and water level is below the required RPV pressure line (301-500 psig – magenta).

d. Correct – This combination of Suppression Pool temperature and water level is above the required RPV pressure line (701-900 psig – blue), therefore HCTL is violated and an Emergency RPV Depressurization is required.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC25 Cert #51 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295036 Secondary Containment High Sump/Area Water Level / 5

2.4.21 - 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.

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295036	
Importance Rating	4.0	

Proposed Question: **# 64**

Unit 1 is operating at 100% power with the following:

- The watertight door between HPCI and RCIC was inadvertently left open.
- A fire header rupture occurs in the HPCI room.
- AR-114-H03, HPCI Pump Room Flooded, alarms.
- AR-108-H03, RCIC Pump Room Flooded, alarms.
- Attempts to isolate the rupture have failed.
- An Operator reports:
  - HPCI room water level is approximately 30", up slow.
  - RCIC room water level is approximately 24", up slow.

Given the following table from EO-000-104, Secondary Containment Control:

TABLE 10 TABLE 10 REACTOR BUILDING WATER LEVEL

RB AREA (645 FT EL)	MAX NORMAL WATER LVL	MAX SAFE WATER LVL (IN)	RB WATER LVL (IN.)
HPCI EQUIPMENT AREA	HI ALARM	27	______
RCIC EQUIPMENT AREA	HI ALARM	23	______
RHR PUMP ROOM A	HI ALARM	84	______
RHR PUMP ROOM B	HI ALARM	85	______
CS PUMP ROOM A	HI ALARM	23	______
CS PUMP ROOM B	HI ALARM	23	______
RB SUMP ROOM	HI ALARM	23	______

Which one of the following describes the required control of the Reactor, in accordance with EO-000-104?

A. Continued Reactor operation is allowed because this is a non-primary leak.

B. Shut down the Reactor in accordance with GO-100-004.

C. Scram the Reactor and rapidly depressurize with Turbine Bypass Valves.

D. Scram the Reactor and perform an Emergency RPV Depressurization.

Proposed Answer: **B**

Explanation:

a. Incorrect – A Reactor shutdown is required. While this is a non-primary leak, which prevents the need for Emergency RPV Depressurization, it does not prevent the need for a Reactor shutdown. This would be the correct answer if only 1 area had exceeded max safe water level.

b. Correct – Two areas have exceeded the max safe water level due to a non-primary system discharge. This requires shutting down the Reactor in accordance with GO-100-004.

c. Incorrect – A Reactor shutdown is required, but a Reactor scram and rapid depressurization are not required since this is a non-primary system discharge. This would be the correct answer if 1 area had exceeded max safe water level, a 2nd area was approaching max safe water level, and the discharge was from a primary system.

d. Incorrect – A Reactor shutdown is required, but a Reactor scram and Emergency RPV Depressurization are not required since this is a non-primary system discharge. This would be the correct answer if this was a primary system leak.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-104 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 NRC #27 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295010 High Drywell Pressure / 5

2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295010	
Importance Rating	4.0	

Proposed Question: **# 65**

Unit 1 is in Mode 3 with the following:

- A dual seal failure occurred on Reactor Recirculation pump (RRP) 1A.
- RRP 1A has been tripped and isolated.
- Drywell pressure is 2.2 psig, steady.
- Drywell average temperature is 145°F, steady.

Which one of the following describes the status of the Limiting Condition for Operation (LCO) for Technical Specifications 3.6.1.4, Containment Pressure, and 3.6.1.5, Drywell Air Temperature?

	LCO 3.6.1.4, Containment Pressure	LCO 3.6.1.5, Drywell Air Temperature
A.	NOT exceeded	NOT exceeded
B.	NOT exceeded	Exceeded
C.	Exceeded	NOT exceeded
D.	Exceeded	Exceeded

Proposed Answer:	D

Explanation:

a. Incorrect – Both limits have been exceeded.

b. Incorrect – LCO 3.6.1.4 has been exceeded.

c. Incorrect – LCO 3.6.1.5 has been exceeded.

d. Correct – Containment pressure is above the limit of 2.0 psig in LCO 3.6.1.4. Drywell average temperature is above the limit of 135°F in LCO 3.6.1.5. Both of these limits are applicable in Modes 1, 2, and 3.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TS 3.6.1.4, TS 3.6.1.5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:
Conduct of Operations

2.1.18 Ability to make accurate, clear and concise logs, records, status boards, and reports.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.18	
Importance Rating	3.6	

Proposed Question: **# 66**

Unit 1 is operating at 100% power with the following:

- You are on day shift as PCOM.
- The following events occur during your shift:

(1) An unexpected alarm is received due to an HCU low accumulator pressure. The accumulator is charged and the alarm is cleared.

(2) A planned swap of the RBCCW pumps is performed.

Which one of the following describes the requirement to log these events in the Unit Log, in accordance with OP-AD-002, Standards for Shift Operations?

A. Neither event must be logged.

B. Event (1) must be logged, only.

C. Event (2) must be logged, only.

D. Both events must be logged.

Proposed Answer:	D

Explanation:

a. Incorrect – Both of these events meet criteria listed in OP-AD-002 Attachment H requiring a log entry.

b. Incorrect – Both of these events meet criteria listed in OP-AD-002 Attachment H requiring a log entry.

c. Incorrect – Both of these events meet criteria listed in OP-AD-002 Attachment H requiring a log entry.

d. Correct – Both of these events meet criteria listed in OP-AD-002 Attachment H requiring a log entry. Criteria 3 requires logging "Starting and stopping equipment controlled from the Main Control Room", which applies to Event (2). Criteria 6 requires logging "Occurrence of significant unexpected or nuisance annunciator alarms with the action taken", which applies to Event (1).

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-AD-002 Attachment H (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
Conduct of Operations	Tier #	3	
	Group #		
2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	2.1.7	
	Importance Rating	4.4	

Proposed Question: **# 67**

Unit 1 has experienced a transient with the following:

- Reactor pressure is 150 psig, down slow.
- 3 SRVs are open.
- Reactor water level is -200", steady.
- Core Spray loop A is injecting at 6420 gpm.

Which one of the following describes the status of core cooling, in accordance with EO-000-102, RPV Control?

A. There is NO assurance of adequate core cooling.

B. Core Spray flow alone ensures adequate core cooling.

C. Reactor water level alone ensures adequate core cooling.

D. The combination of Core Spray flow and Reactor water level ensures adequate core cooling.

Proposed Answer:	D

Explanation:

a. Incorrect – The combination of Core Spray flow (>6350 gpm) and Reactor water level (>-210") provides adequate core cooling.

b. Incorrect – While Core Spray flow is above the design requirement (6350 gpm), this is not enough to ensure adequate core cooling alone. Reactor water level also must be >-210".

c. Incorrect – With injection to the Reactor, Reactor water level would need to be >-179" to ensure adequate core cooling alone. Reactor water level >-205" only assures adequate core cooling if there is no injection to the Reactor.

d. Correct – The combination of Core Spray flow (>6350 gpm) and Reactor water level (>-210") provides adequate core cooling.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 NRC #16

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC24 NRC #16

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:
Equipment Control

2.2.13 - Knowledge of tagging and clearance procedures.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.2.13	
Importance Rating	4.1	

Proposed Question: **# 68**

A tagout is being developed for a water system with the following conditions:

- Highest system water temperature: 250°F
- Highest system water pressure: 250 psig

Which one of the following describes if the system is considered a "High Energy System", in accordance with NDAP-QA-0322, Energy Control Process?

The system is...

A. NOT a "High Energy System".

B. a "High Energy System" due to pressure only.

C. a "High Energy System" due to temperature only.

D. a "High Energy System" due to both pressure and temperature.

Proposed Answer:	**C**

Explanation:

a. Incorrect – The system is considered a "High Energy System" because water temperature is >200°F.

b. Incorrect – The system pressure is below the 800 psig threshold for a "High Energy System", however the system temperature is above the 200°F threshold.

c. Correct – The system is considered a "High Energy System" because water temperature is >200°F. System pressure is below the 800 psig threshold for a "High Energy System".

d. Incorrect – System pressure is below the 800 psig threshold for a "High Energy System".

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Technical Reference(s): NDAP-QA-0322 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # JAF 9/14 NRC #67

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam JAF 9/14 NRC #67

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:
Equipment Control

2.2.3 Knowledge of the design, procedural, and operational differences between units.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.2.3	
Importance Rating	3.8	

Proposed Question: **# 69**

Which one of the following describes a difference between the required actions of EO-100-030 and EO-200-030, the Station Blackout procedures for Unit 1 and Unit 2, respectively?

A. Unit 1 is required to align Blue Max to DC battery chargers, while Unit 2 is NOT.

B. Unit 2 is required to align Blue Max to DC battery chargers, while Unit 1 is NOT.

C. Unit 1 is required to secure numerous emergency DC lube oil pumps, while Unit 2 is NOT.

D. Unit 2 is required to secure numerous emergency DC lube oil pumps, while Unit 1 is NOT.

Proposed Answer:	**C**

Explanation:

a. Incorrect – Each Unit may or may not align Blue Max to DC batter chargers, depending on which DC power restoration strategy is employed.

b. Incorrect – Each Unit may or may not align Blue Max to DC batter chargers, depending on which DC power restoration strategy is employed.

c. Correct – Due to difference in load shedding requirements, Unit 1 is required to secure numerous emergency DC lube oil pumps in 30 to 45 minutes of the start of a Station Blackout, while Unit 2 is NOT required to take these same actions.

d. Incorrect – Unit 1 is required to secure numerous emergency DC lube oil pumps, NOT Unit 2.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-100-030, EO-200-030 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 NRC #70 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #	3	
	Group #		
2.3.13 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	K/A #	2.3.13	
	Importance Rating	3.4	

Proposed Question: **# 70**

Unit 1 is operating at 3% power with the following:

- Maintenance personnel have entered the Drywell to perform emergent repairs on elevation 719'.
- The PCOM notes that Reactor power is rising unexpectedly.

Which one of the following actions must the PCOM take, in accordance with NDAP-QA-0309, Primary Containment Access and Control?

A. Place the Reactor Mode Switch to SHUTDOWN.

B. Notify HP to resurvey the 719' elevation of the Drywell

C. Direct personnel to move down to Drywell elevation 704'.

D. Manually insert control rods with RMCS to maintain power <3%.

Proposed Answer:	A

Explanation:

a. Correct – NDAP-QA-0309 section 6.5 provides guidance for control of Reactor power during Drywell entries with the Reactor operating. This procedures requires the PCOM to initiate a Reactor scram by placing the Reactor Mode Switch to SHUTDOWN on any unexpected power rise.

b. Incorrect – Notifying HP to resurvey the Drywell would be an appropriate follow-up action, but is not the required action to deal with the immediate consequences of the unexpected power rise.

c. Incorrect – With power unexpectedly rising, the required action is to place the Reactor Mode Switch in SHUTDOWN, not relocate personnel.

d. Incorrect – The procedure specifically requires placing the Reactor Mode Switch in SHUTDOWN, not taking slower actions in an attempt to lower Reactor power, since the power rise is unexpected. Control rod insertion with RMCS would be allowed if power were <3% and the power rise was expected (such as from Xenon burnout).

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LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): NDAP-QA-0309 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC26 NRC #70

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC26 NRC #70

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #	3	
2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	Group #		
	K/A #	2.3.4	
	Importance Rating	3.2	

Proposed Question: **# 71**

Unit 1 is shutdown with the following:

- A 31 year old Operator is entering the Drywell for a job.
- Expected dose for transiting to and from the job location is a total of 20 mRem.
- Expected dose rate at the job location is 3 Rem/hr.
- It will take 45 minutes to complete the job.
- The Operator's previous TEDE for the year is 1630 mRem.
- No dose extension has been previously obtained for this Operator.
- No emergency is in progress.

Which one of the following describes the Operator's expected dose, in accordance with NDAP-QA-0625, Personnel Radiation Exposure Monitoring Program?

The Operator's expected dose...

A. will stay within the normal annual dose control level *without* an extension.

B. will exceed the normal annual dose control level *without* an extension, but stay within the allowable dose control level *with* an extension.

C. will exceed the allowable dose control level *with* an extension, but stay within the federal dose limit.

D. will exceed the federal dose limits.

Proposed Answer:	**B**

Explanation:

a. Incorrect – The Operator's annual dose will be 3900 mRem, which is above the normal annual dose control level without an extension (2000 mRem).

b. Correct – The given job information will result in the Operator's annual dose rising to 3900 mRem (1630 mRem + 20 mRem + .75 hours * 3000 mRem/hour). This is above the normal annual dose control level without an extension (2000 mRem), but below the maximum allowable dose with an extension (4000 mRem).

c. Incorrect – The Operator's annual dose will be 3900 mRem, which is below the maximum allowable dose with an extension (4000 mRem).

d. Incorrect – The Operator's annual dose will be 3900 mRem, which is below the federal dose limit (5000 mRem).

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): NDAP-QA-0625 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # JAF 4/14 NRC #75

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam JAF 4/14 NRC #75

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 12

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Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:
Radiation Control

2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.5	
Importance Rating	2.9	

Proposed Question: **# 72**

Which one of the following describes:

(1) The indicating light(s) available on an Area Radiation Monitor (ARM) indicator and trip unit on Panel 1C605.

(2) How the light(s) respond from a normal steady state condition when an upscale condition occurs and then subsequently ***clears***?

	Indicating Light(s) Available	Light(s) Response When Initiating Condition Clears
A.	Amber alarm light, only	Automatically extinguishes
B.	Amber alarm light, only	Remains illuminated until Operator resets unit
C.	Amber alarm light and white downscale light	Automatically extinguishes
D.	Amber alarm light and white downscale light	Remains illuminated until Operator resets unit

Proposed Answer:	**D**

Explanation:

a. Incorrect – The ARM indicator and trip unit on Panel 1C605 has both an amber alarm light and a white downscale light. The local alarm unit has only the amber alarm light.

b. Incorrect – The ARM indicator and trip unit on Panel 1C605 has both an amber alarm light and a white downscale light. The local alarm unit has only the amber alarm light.

c. Incorrect – These lights seal-in upon receipt of an initiating condition. When the initiating condition clears, the associated light remains illuminated until Operator action is taken to manually reset the unit.

d. Correct – The ARM indicator and trip unit on Panel 1C605 has both an amber alarm light and a white downscale light. The local alarm unit has only the amber alarm light. These lights seal-in upon receipt of an initiating condition. When the initiating condition clears, the associated light remains illuminated until Operator action is taken to manually reset the unit.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TM-OP-079B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 NRC #71

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam LOC24 NRC #71

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures/Plan	Tier #	3	
2.4.3 Ability to identify post-accident instrumentation.	Group #		
	K/A #	2.4.3	
	Importance Rating	3.7	

Proposed Question: **# 73**

A Control Room evacuation has been performed and the laptops at the Remote Shutdown Panels are NOT available.

Given the following plant parameters:

(1) Drywell pressure
(2) APRM power level
(3) Suppression Pool water temperature

Which one of the following identifies which of these parameters have installed indicators at the Remote Shutdown Panels?

A. (1) and (2), only

B. (1) and (3), only

C. (2) and (3), only

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

Proposed Answer:	**B**

Explanation:

a. Incorrect – The Remote Shutdown Panels have installed indicators for Suppression Pool water temperature, but not for APRM power level.

b. Correct – The Remote Shutdown Panels have installed indicators for both Drywell pressure and Suppression Pool water temperature, but not for APRM power level.

c. Incorrect – The Remote Shutdown Panels have installed indicators for Drywell pressure, but not for APRM power level.

d. Incorrect – The Remote Shutdown Panels do not have installed indicators for APRM power level.

Technical Reference(s): TM-OP-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis ____

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55.43 ____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Emergency Procedures/Plan	Tier #	3	
2.4.18 Knowledge of the specific bases for EOPs.	Group #		
	K/A #	2.4.18	
	Importance Rating	3.3	

Proposed Question: **# 74**

Which one of the following describes the basis for maintaining plant parameters within the Heat Capacity Temperature Limit, in accordance with EO-1(2)00-103, Primary Containment Control?

Ensure that...

A. Suppression Chamber pressure stays within the Pressure Suppression Limit.

B. Core Spray and RHR suction piping design temperature limits are NOT exceeded.

C. Emergency Core Cooling system net positive suction head limits are NOT violated.

D. Primary Containment vent valve operability is maintained following RPV Depressurization.

Proposed Answer:	**D**

Explanation:

a. Incorrect – PSL is a lower limit on Suppression Chamber pressure compared to the 65 psig associated with HCTL (operability limit for Primary Containment vent valves). However, PSL is not the basis for maintaining plant parameters below HCTL.

b. Incorrect – High Suppression Pool water temperature does raise Core Spray and RHR suction temperature, but this is not the basis for maintaining Suppression Pool water temperature below HCTL.

c. Incorrect – High Suppression Pool water temperature does reduce available NPSH for multiple ECCS pumps, but this is not the basis for maintaining Suppression Pool water temperature below HCTL.

d. Correct – EO-000-103 states, "If RPV pressure, suppression pool temperature and suppression pool level cannot be maintained below the HCTL, the primary containment vent valve opening pressure may be exceeded following RPV depressurization."

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP1 2015 Audit #64

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:
Emergency Procedures/Plan

2.4.37 Knowledge of the lines of authority during implementation of the emergency plan.

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.4.37	
Importance Rating	3.0	

Proposed Question: **# 75**

An emergency has been declared with the following:

- All Emergency Response Facilities have been activated.
- An In-Plant (India) Team is required to be dispatched to the field to isolate a leak.

Which one of the following identifies which facility is responsible for assembling, briefing, and dispatching the In-Plant (India) Team?

A. Main Control Room

B. Technical Support Center (TSC)

C. Operations Support Center (OSC)

D. Emergency Operations Facility (EOF)

Proposed Answer:	C

Explanation:

a. Incorrect – During non-emergency conditions, the Control Room normally is responsible for dispatching personnel for off-normal response in the plant. However, during a declared emergency with all Emergency Response Facilities activated, this responsibility moves to the Operations Support Center.

b. Incorrect – The Technical Support Center is involved in determining priorities for emergency response actions, but is not directly responsible for the actual assembling, briefing, and dispatching of In-Plant (India) Teams.

c. Correct – The Operation Support Center is responsible for assembling, briefing, and dispatching In-Plant (India) Teams during a declared emergency.

d. Incorrect – The Emergency Operations Facility maintains overall oversight of the emergency response, but is not directly responsible for the actual assembling, briefing, and dispatching of In-Plant (India) Teams.

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REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EP-PS-100 Tab B Item 18c, EP-PS-132 Page 2 and Tab D (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

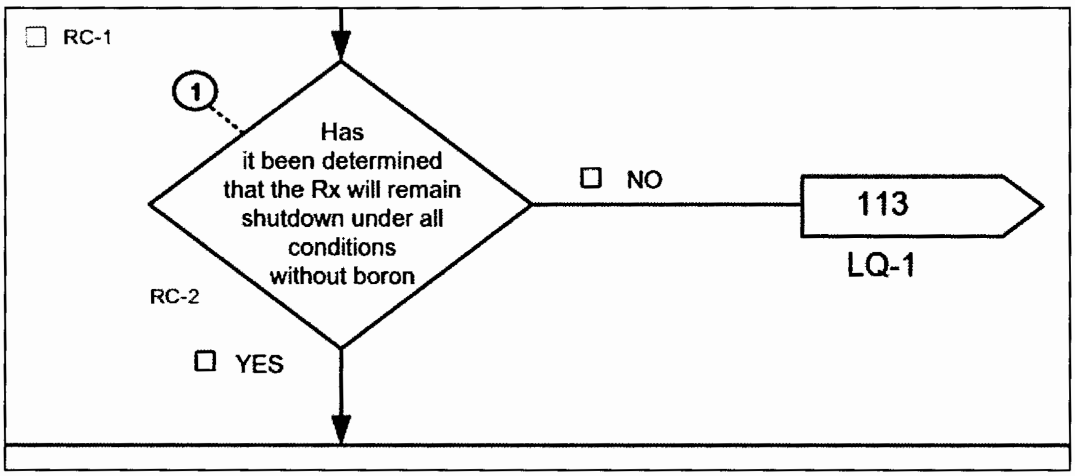
SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295006 SCRAM / 1	Tier #		1
AA2.05 - Ability to determine and/or interpret the following as they apply to SCRAM: Whether a reactor SCRAM has occurred	Group #		1
	K/A #	295006	
	Importance Rating	4.6	

Proposed Question: **# 76**

Unit 1 has been scrammed due to low Condenser vacuum with the following:

- APRMs indicate downscale.
- IRMs are fully inserted and indicate downscale on Range 1.
- All SRMs are unavailable.
- Ten control rods remain withdrawn at position 04.
- All other control rods are fully inserted.
- EO-000-102, RPV Control, has been entered due to low Reactor water level.
- The next step in EO-000-102 is:



Which one of the following describes how this step is required to be answered, in accordance with EO-000-102?

Answer...

A. YES based on APRM and IRM indications.

B. YES based on control rod positions.

C. NO based on lack of SRM indications.

D. NO based on control rod positions.

Proposed Answer: **D**

Explanation:

a. Incorrect – Although the given APRM and IRM indications show that the scram has resulted in a significant reduction in Reactor power, the given control rod positions prevent the SRO from answering this question YES because more than one control rod is withdrawn past position 00 (the Maximum Subcritical Banked Withdrawal Position at SSES). The given power indications show that the Reactor is currently shutdown, but not that the Reactor will stay shutdown under all conditions without boron, as required in this step.

b. Incorrect – Although all control rods have inserted to at least position 04, this is not enough control rod insertion for the SRO to answer this question YES. This question can only be answered YES if one control rod remains withdrawn past position 00 (the Maximum Subcritical Banked Withdrawal Position at SSES).

c. Incorrect – This question is required to be answered NO, but based on control rod positions, not the lack of SRM indication. SRM indication is helpful in determining Reactor status, but are not required for answering this step.

d. Correct – This question is required to be answered NO because more than one control rod is withdrawn past position 00 (the Maximum Subcritical Banked Withdrawal Position at SSES).

Note: The K/A requires the question to test ability to determine/interpret whether a Reactor scram has occurred. There are multiple aspects of determining/interpreting whether a Reactor scram has occurred (such as RPS logic status, CRD response, fulfillment of reactivity control safety function). In order to test at an appropriate SRO level, this question is testing the ability of the SRO to determine if a scram has resulted in fulfillment of the reactivity control safety function (i.e. will the Reactor stay shutdown under all conditions without boron?) through proper execution of a decision in the EOPs.

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s):	EO-000-102	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295005 Main Turbine Generator Trip / 3

AA2.04 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor pressure

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295005	
Importance Rating	3.8	

Proposed Question: **# 77**

Unit 1 is operating at 100% power with the following:

- Engineering and Maintenance are performing inspections in preparation for an upcoming outage.
- From these inspections, it is determined that two (2) of the Turbine Bypass Valves (TBVs) are inoperable and unable to perform their pressure control function on a Main Turbine trip.

Which one of the following describes how this finding affects Technical Specifications?

Technical Specification 3.7.6, Main Turbine Bypass System, LCO is...

A. still met by the three (3) remaining operable TBVs.

B. NOT met by the three (3) remaining operable TBVs. The Reactor must be shutdown.

C. NOT met by the three (3) remaining operable TBVs. Reactor operation may continue, but only if power is lowered to less than 23%.

D. NOT met by the three (3) remaining operable TBVs. Reactor operation may continue at the current power level if thermal limit penalties are satisfied.

Proposed Answer:	**D**

Explanation:

a. Incorrect – The COLR requires at least 4 operable TBVs to satisfy TS 3.7.6 LCO. With only 3 operable TBVs, TS 3.7.6 LCO is NOT met.

b. Incorrect – TS 3.7.6 LCO is NOT met, however Reactor operation may continue as long as either the applicable thermal limit penalties are satisfied or Reactor power is <23%.

c. Incorrect – TS 3.7.6 LCO is NOT met, however Reactor operation may continue at the current Reactor power level as long as the applicable thermal limit penalties are satisfied.

d. Correct – The COLR requires at least 4 operable TBVs to satisfy TS 3.7.6 LCO. With only 3 operable TBVs, TS 3.7.6 LCO is NOT met by the number of operable TBVs. TS 3.7.6 still allows Reactor operation at the current power level as long as the applicable thermal limit penalties are satisfied.

Technical Reference(s): TS 3.7.6 and bases, COLR (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # NMP1 2013 Audit #84 (Note changes or attach parent)

New

Question History: Last NRC Exam

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
295028 High Drywell Temperature / 5	Tier #		1
EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level	Group #		1
	K/A #	295028	
	Importance Rating	3.9	

Proposed Question: **# 78**

Unit 1 was operating at 100% power when a significant event resulted in the following:

- Reactor pressure is 75 psig, down slow.
- Drywell temperature is 350°F, down slow.
- Instrument run temperature on UR-15701A is 350°F, down slow.
- Narrow range Reactor water level instruments are indicating 0", steady.
- Wide range Reactor water level instruments are indicating -85", steady.
- Fuel Zone Reactor water level instruments are indicating upscale.
- Shutdown and Upset range Reactor water level instruments are indicating downscale.
- Extended range Reactor water level instruments are indicating erratically.

Note: Portions of EO-000-102, RPV Control, and EO-000-103, Primary Containment Control, are provided on the following page.

Which one of the following describes the need to enter EO-000-114, RPV Flooding Non-ATWS, in accordance with EO-000-102?

Entry into EO-000-114 is...

A. required because Reactor water level can NOT be determined using any instruments.

B. required because Reactor water level can only be determined by one range of instruments.

C. NOT required because Reactor water level can be determined using Wide range instruments.

D. NOT required because Reactor water level can be determined using Narrow range instruments.

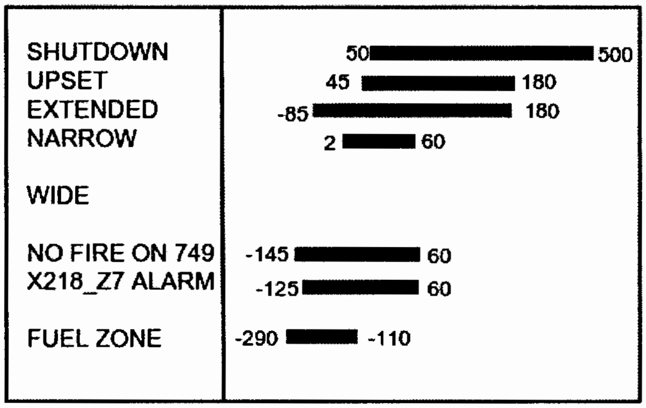
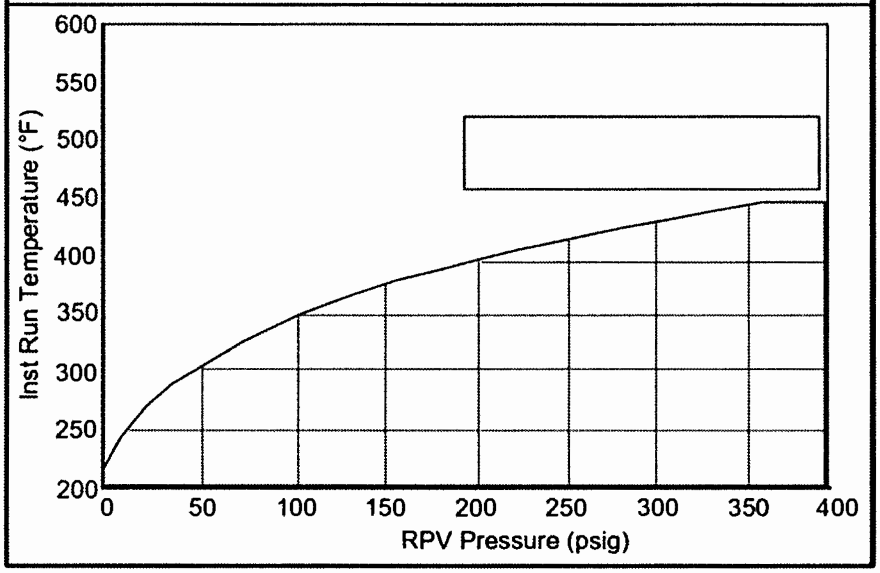


FIG. 1
RPV SATURATION CURVE



Proposed Answer:	C

Explanation:

a. Incorrect – Wide range instruments are indicating in their usable range and are not showing signs of flashing, therefore they accurately indicate Reactor water level. This is true even though the given combination of instrument run temperature and Reactor pressure is above the RPV Saturation Curve. Since Reactor water level can be determined, entry into EO-000-114 is not required.

b. Incorrect – Wide range instruments are indicating in their usable range and are not showing signs of flashing, therefore they accurately indicate Reactor water level. Since Reactor water level can be determined, entry into EO-000-114 is not required. While use of redundant instruments is an important practice, there are multiple Wide range instruments and there is no requirement for another range of Reactor water level instruments to be indicating to avoid entry into EO-000-114.

c. Correct - Wide range instruments are indicating in their usable range of -145" to 60". This range is valid since instrument run temperature is not above 350°F. The combination of instrument run temperature and Reactor pressure are exceeding the RPV Saturation Curve, which indicates that flashing could be of concern. However, Wide range instruments are indicating steady, so no sign of flashing is present. Therefore, Wide range instruments accurately indicate Reactor water level. Since Reactor water level can be determined, entry into EO-000-114 is not required.

d. Incorrect – Narrow range instruments are below their minimum usable level of +2". They therefore cannot be used to determine Reactor water level with respect to the top of active fuel, and are inadequate by themselves to prevent entry into EO100-114.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EO-000-102, EO-000-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Question Source: Bank #

Modified Bank # JAF 9/14 NRC #57 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-site Release Rate / 9	Tier #		1
2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures	Group #		1
	K/A #	295038	
	Importance Rating	4.2	

Proposed Question: **# 79**

Unit 1 is operating at 80% power during a power ascension with the following:

- AR-015-E04, STACK MONITORING SYS 0C630/0C677 HI RADIATION, alarms.
- AR-015-D04, STACK MONITORING SYS 0C630/0C677 HI-HI RADIATION, alarms.
- Turbine Building Exhaust Radiation (Point #5) is the cause of the alarms.
- Offgas subtrain flow is 75% of the value it was before the power ascension began.
- Offgas recombiner flow has risen as power has risen.

Which one of the following is required?

A. Enter ON-070-001, Abnormal Gaseous Release / CAM Alarms, and direct a Reactor shutdown.

B. Enter ON-070-001, Abnormal Gaseous Release / CAM Alarms, and direct investigation of Offgas Delay Line Drain valves.

C. Enter ON-179-001, Increasing Offgas / MSL Rad Levels, and direct a Reactor power reduction per ON-RPR-102, Rapid Power Reduction.

D. Enter ON-179-001, Increasing Offgas / MSL Rad Levels, and direct a Reactor scram per ON-SCRAM-101, Reactor Scram.

Proposed Answer:	B

Explanation:

a. Incorrect – ON-070-001 entry is required based on the provided alarms and associated TB Exhaust Radiation reading. However, these conditions do not yet require a Reactor shutdown. A Reactor shutdown would only be required if conditions degraded to the point where EO-000-105, Radioactivity Release Control, entry was required and offsite release rate was threatening public health and safety.

b. Correct - ON-070-001 entry is required based on the provided alarms and associated TB Exhaust Radiation reading. The high Turbine Building Exhaust Radiation combined with the abnormal Offgas flow indication suggests the Offgas Delay Line Drain valves may be improperly aligned. ON-070-001 requires investigation of the valves in these conditions.

c. Incorrect – While rising MSL / Offgas radiation levels must be investigated as part of ON-070-001, the given indications do not yet require entry into ON-179-001. The given indications suggest the Offgas Delay Line Drain valves may be improperly aligned. ON-179-001 does contain steps to perform a Reactor power reduction to attempt to stabilize / limit offsite release.

d. Incorrect – While rising MSL / Offgas radiation levels must be investigated as part of ON-070-001, the given indications do not yet require entry into ON-179-001. The given indications suggest the Offgas Delay Line Drain valves may be improperly aligned. ON-179-001 does contain steps that require a Reactor scram if conditions exceed the MSL Hi-Hi rad level, but not the Stack Hi-Hi rad level.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): AR-015-D04, ON-070-001, ON-179-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 Cert #83

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
700000 Generator Voltage and Electric Grid Disturbances	Tier #		1
2.2.25 - Equipment Control: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	Group #		1
	K/A #	700000	
	Importance Rating	4.2	

Proposed Question: **# 80**

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

- Thunderstorms in the area have caused offsite grid disturbances.
- The load tap changer for Startup Transformer 20 has failed to respond to changing grid conditions.
- Investigation reveals that the load tap changer for Startup Transformer 20 is stuck and can NOT be moved.
- Startup Bus 20 voltage is 14 kV.
- All 4160 V ESS Bus voltages are 4200 V.

Which one of the following describes the operability of the offsite power sources and the 4160 V ESS Buses, in accordance with Technical Specifications?

A. All offsite power sources and 4160 V ESS Buses remain operable.

B. One offsite power source is inoperable. All 4160 V ESS Buses remain operable.

C. Half of the 4160 V ESS Buses are inoperable. All offsite power sources remain operable.

D. One offsite power source is inoperable and half of the 4160 V ESS Buses are inoperable

Proposed Answer:	B

Explanation:

a. Incorrect - TS 3.8.1 Bases state that for an offsite power source to be operable the associated transformer load tap transformer must be operating properly in automatic. With the load tap changer for Startup Transformer 20 stuck, one offsite power source must be declared inoperable.

b. Correct – TS 3.8.1 Bases state that for an offsite power source to be operable the associated transformer load tap transformer must be operating properly in automatic. With the load tap changer for Startup Transformer 20 stuck, one offsite power source must be declared inoperable. TS 3.8.7 Bases do not require this load tap changer to be operating properly to support operability of the 4160 V ESS Buses. Since voltage is still normal on these buses, they remain operable.

c. Incorrect - TS 3.8.1 Bases state that for an offsite power source to be operable the associated transformer load tap transformer must be operating properly in automatic. With the load tap changer for Startup Transformer 20 stuck, one offsite power source must be declared inoperable. TS 3.8.7 Bases do not require this load tap changer to be operating properly to support operability of the 4160 V ESS Buses. Since voltage is still normal on these buses, they remain operable.

d. Incorrect - TS 3.8.7 Bases do not require this load tap changer to be operating properly to support operability of the 4160 V ESS Buses. Since voltage is still normal on these buses, they remain operable.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TS 3.8.1 and bases, TS 3.8.7 and bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # Vision SYSID 34623

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

2.4.9 - Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295001	
Importance Rating	4.2	

Proposed Question: **# 81**

Unit 1 is cooling down with the following:

- Reactor pressure is 30 psig, down slow.
- RHR loop A is operating in the Shutdown Cooling lineup.
- Both Reactor Recirculation pumps are secured.

Then, a leak results in the following:

- The crew isolates the Shutdown Cooling lineup.
- Reactor water level stabilizes at 35".
- Shutdown Cooling CANNOT be quickly restored.

Which one of the following describes the status of Reactor coolant circulation, in accordance with ON-SDC-101, Loss of Shutdown Cooling?

Reactor coolant circulation is...

A. adequate with the current conditions.

B. inadequate. ONLY reactor water level must be raised.

C. inadequate. ONLY a Reactor Recirculation pump must be started.

D. inadequate. Either Reactor water level must be raised or a Reactor Recirculation pump must be started.

Proposed Answer: **D**

Explanation:

a. Incorrect – With Reactor water level <45", SDC out of service, and no RRP in operation, Reactor coolant circulation is inadequate. If Reactor water level were ≥45", then adequate Reactor coolant circulation would exist even with no SDC or Recirc pumps operating.

b. Incorrect – Reactor water level does not have to be raised. Starting a RRP is another valid option to restore adequate Reactor coolant circulation.

c. Incorrect – Starting a RRP is not required. Raising Reactor water level is another valid option to restore adequate Reactor coolant circulation.

d. Correct – With Reactor water level <45", SDC out of service, and no RRP in operation, Reactor coolant circulation is inadequate.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-SDC-101 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

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
295004 Partial or Total Loss of DC Pwr / 6	Tier #		1
2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions.	Group #		1
	K/A #	295004	
	Importance Rating	4.4	

Proposed Question: **# 82**

Unit 1 is operating at 100% power with the following:

- Annunciator AR-106-B12, 125V DC Panel 1L620 System Trouble, alarms.
- The supply breaker to 1D624 is tripped and can NOT be reset.

Which one of the following describes the required actions, in accordance with Technical Specifications?

Re-energize the load group within a maximum of __(1)__ hours, or then __(2)__.

	(1)	(2)
A.	2	be in at least Hot Shutdown within the next 12 hours
B.	8	be in at least Hot Shutdown within the next 12 hours
C.	2	transfer the Unit 1 and Common loads to the Unit 2 load groups within the next 12 hours to avoid the need for a shutdown
D.	8	transfer the Unit 1 and Common loads to the Unit 2 load groups within the next 12 hours to avoid the need for a shutdown

Proposed Answer:	**A**

Explanation:

a. Correct – 1D624 is one of the DC subsystems required by Technical Specification 3.8.7. With 1D624 de-energized, Condition B must be entered. Required Action B.1 requires re-energizing 1D624 within a maximum of 2 hours or then Condition C must be entered. Required Action C.1 requires being in at least Mode 3 (Hot Shutdown) within the next 12 hours.

b. Incorrect – 8 hours is the time limit in Technical Specification 3.8.7 for AC subsystems, but DC subsystems require a more restrictive 2 hour time limit.

c. Incorrect – Many loads on 1D624 may be transferred to Unit 2 and would be per ON-125VDC-101. However, Technical Specification 3.8.7 does not provide for this as a method to avoid the required shutdown of Condition C.

d. Incorrect – 8 hours is the time limit in Technical Specification 3.8.7 for AC subsystems, but DC subsystems require a more restrictive 2 hour time limit. Many loads on 1D624 may be transferred to Unit 2 and would be per ON-125VDC-101. However, Technical Specification 3.8.7 does not provide for this as a method to avoid the required shutdown of Condition C.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): AR-106-B12, ON-102-620, ON-125VDC-101 TS 3.8.7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TS 3.8.7 w/o bases

Question Source: Bank # Vision SYSID 5578

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Water Level / 5	Tier #		1
EA2.01 - Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Suppression pool water level	Group #		2
	K/A #	295029	
	Importance Rating	3.9	

Proposed Question: **# 83**

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is 0", up slow.
- Reactor pressure is 200 psig, down slow.
- Drywell pressure is 10 psig, down slow.
- Suppression Pool water level is 40.5', up slow.
- Condensate is injecting to the Reactor.
- Core Spray and RHR are available for injection.
- RHR loop A is spraying both the Drywell and Suppression Chamber.

Note: A portion of EO-000-103, Primary Containment Control, is provided on the following page.

Which one of the following describes the required control of Reactor injection and Drywell spray, in accordance with EO-000-103?

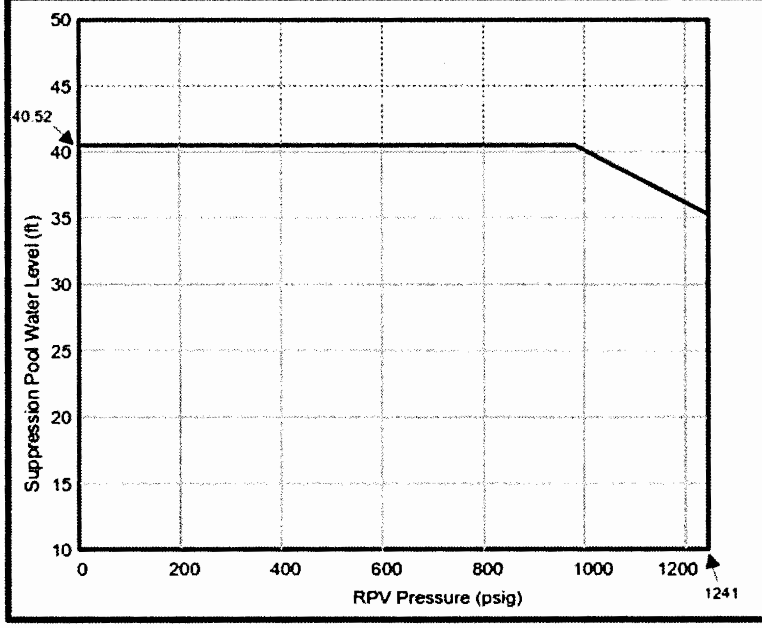
A. Continued injection with Condensate and operation of Drywell spray is acceptable.

B. Condensate injection must be terminated. Continued operation of Drywell spray is acceptable.

C. Continued injection with Condensate is acceptable. Drywell spray must be terminated.

D. Condensate injection and Drywell spray must be terminated.

FIG.3 STPLL
SRV TAILPIPE LEVEL LIMIT



Proposed Answer: **B**

Explanation:

a. Incorrect – With SP water level at 40.5' and still rising, the plant cannot be maintained within the SRV tailpipe level limit. EO-000-103 step SP/L-13 requires terminating injection in the Reactor from sources external to the Containment if adequate core cooling is assured. Condensate injects water from external to the Containment. Since Reactor water level is well above top of active fuel and alternate injection system are available, Condensate injection must be terminated.

b. Correct – With SP water level at 40.5' and still rising, the plant cannot be maintained within the SRV tailpipe level limit. EO-000-103 step SP/L-13 requires terminating injection in the Reactor from sources external to the Containment if adequate core cooling is assured. Condensate injects water from external to the Containment. Since Reactor water level is well above top of active fuel and alternate injection system are available, Condensate injection must be terminated. EO-000-103 step SP/L-17 does not require stopping Drywell sprays unless SP level cannot be maintained below 43'. Additionally, this step is not evaluated until after Condensate injection is secured. Securing condensate injection and starting injection with Core Spray and/or RHR will likely prevent approaching 43'. Therefore, Drywell sprays may remain in operation.

c. Incorrect – With SP water level at 40.5' and still rising, the plant cannot be maintained within the SRV tailpipe level limit. EO-000-103 step SP/L-13 requires terminating injection in the Reactor from sources external to the Containment if adequate core cooling is assured. Condensate injects water from external to the Containment. Since Reactor water level is well above top of active fuel and alternate injection system are available, Condensate injection must be terminated.

d. Incorrect - EO-000-103 step SP/L-17 does not require stopping Drywell sprays unless SP level cannot be maintained below 43'. Additionally, this step is not evaluated until after Condensate injection is secured. Securing condensate injection and starting injection with Core Spray and/or RHR will likely prevent approaching 43'. Therefore, Drywell sprays may remain in operation.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s):	EO-000-103	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #	JAF 9/14 NRC #59	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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
295007 High Reactor Pressure / 3	Tier #		1
2.4.35 - Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.	Group #		2
	K/A #	295007	
	Importance Rating	4.0	

Proposed Question: **# 84**

Unit 1 is operating at 100% power with the following:

- A significant fire develops in the Control Room.
- The Control Room is evacuated ***before*** any of the immediate operator actions are taken.
- Due to evacuating prior to completing the immediate operator actions, the Unit Supervisor directs an Operator to perform the required extra field actions, in accordance with ON-100-009, Control Room Evacuation.
- The Operator completes these required field actions in accordance with ON-100-009.
- NO actions have yet been taken at the Remote Shutdown Panel.

Which one of the following describes the resulting Reactor pressure response, in accordance with ON-100-009?

Reactor pressure is...

A. less than 935 psig, down slow.

B. approximately 935 psig, steady.

C. approximately 1035 psig, steady.

D. greater than 1035 psig, cycling.

Proposed Answer:	D

Explanation:

a. Incorrect – Reactor pressure will be high with SRVs cycling because the directed actions scram the Reactor and closed MSIVs without manually opening any SRVs. Once control is established at the Remote Shutdown Panel, one or more SRVs will be opened and Reactor pressure will be lowered.

b. Incorrect – Reactor pressure will be high with SRVs cycling because the directed actions scram the Reactor and closed MSIVs without manually opening any SRVs. Reactor pressure would be approximately 935 psig, steady, on TBVs if the directed actions caused a Reactor scram but did not close MSIVs.

c. Incorrect – Reactor pressure will be high with SRVs cycling because the directed actions scram the Reactor and closed MSIVs without manually opening any SRVs. Reactor pressure would be approximately 1035 psig if the Reactor were maintained at 100% until later actions in ON-100-109 were performed (such as at the RSP, or after RSP panel actions are taken).

d. Correct – The required field actions in response to an evacuation occurring prior to immediate operator actions being performed in the Control Room result in a Reactor scram and closure of the MSIVs. The actions do not include any manipulation of SRVs until control is established at the RSP. Therefore, at the given point in execution of ON-100-109, Reactor pressure will be >1035 psig and cycling up and down as SRVs actuate in pressure relief mode (~1100 psig).

Note: The question meets SRO level guidelines by requiring detailed knowledge of subsequent actions in ON-100-109. The candidate must know what actions are taken in response to an alternate path in this off-normal procedure, including the ordering of these actions.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-100-109 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New x

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 ____

55.43 5

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

295014 Inadvertent Reactivity Addition / 1

AA2.05 - Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: Violation of safety limits

Level	RO	SRO
Tier #		1
Group #		2
K/A #	295014	
Importance Rating	4.6	

Proposed Question: **# 85**

Unit 1 is operating at 100% power with the following:

- A control rod drifts from position 00 to 48 and can NOT be re-inserted.
- Reactor power reaches a maximum of 102% during the transient.
- Operators restore Reactor power to within rated by lowering Recirculation flow.
- MCPR reached a low of 1.05 during the transient.
- MCPR is now 1.25.

Which one of the following describes the need for a further Reactor power reduction and to make an ENS notification to the NRC, in accordance with Technical Specifications and NDAP-QA-0720, Station Report Matrix and Reportability Evaluation Guidance?

Further Reactor power reduction is ___(1)___. ENS notification to the NRC is ___(2)___.

	(1)	(2)
A.	required	required
B.	required	NOT required
C.	NOT required	required
D.	NOT required	NOT required

Proposed Answer: **A**

Explanation:

a. Correct – MCPR < 1.09, even for a brief period of time, results in violation of Safety Limit 2.1.1.2. This Safety Limit violation requires both restoration of compliance with the limit and inserting all insertable control rods within 2 hours. The need to insert control rods is NOT avoided by restoring compliance with the Safety Limit. Therefore, further power reduction is required to perform a shutdown per Technical Specifications. NDAP-QA-0720 requires a 4 hour ENS notification to the NRC due to a shutdown required by Technical Specifications.

b. Incorrect – Although NDAP-QA-0720 does not specifically include a notification requirement for violation of a Safety Limit, a 4 hour ENS notification to the NRC is required as a result of the ensuing Technical Specification required shutdown.

c. Incorrect – Even though MCPR was quickly restored to above the Safety Limit, all insertable control rods must be inserted within 2 hours.

d. Incorrect – Even though MCPR was quickly restored to above the Safety Limit, all insertable control rods must be inserted within 2 hours. Although NDAP-QA-0720 does not specifically include a notification requirement for violation of a Safety Limit, a 4 hour ENS notification to the NRC is required as a result of the ensuing Technical Specification required shutdown.

<u>Note:</u> The question meets SRO level guidance by requiring the candidate to interpret the effect of a Safety Limit violation on reporting requirements. Solely requiring the candidate to determine that a Safety Limit violation has occurred would not meet SRO level guidance.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): TS Section 2.0, NDAP-QA-0720 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NDAP-QA-0720

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor	Tier #		2
A2.02 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SRM inop condition	Group #		1
	K/A #	215004	
	Importance Rating	3.7	

Proposed Question: **# 86**

Unit 1 is in Mode 5 with the following:

- Irradiated fuel movements in the Reactor core are about to commence.
- There is irradiated fuel in all four (4) quadrants of the Reactor core.
- SRMs are indicating as follows:

SRM	Count Rate (cps)	Signal to Noise Ratio
A	90	6:1
B	0.7	4:1
C	40	5:1
D	10	1:1

Note: A core map showing SRM locations is provided on the following page.

Which one of the following identifies the number of core quadrants in which irradiated fuel bundles may be moved, in accordance with Technical Specifications?

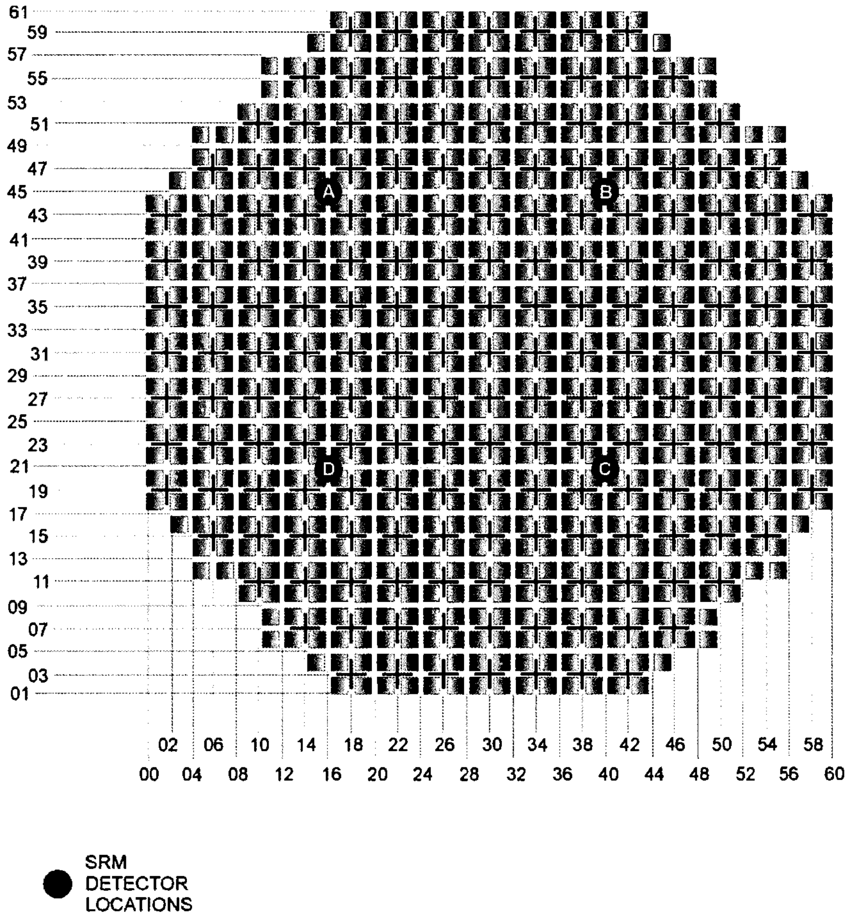
A. 0

B. 2

C. 3

D. 4

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION



Proposed Answer:	**A**

Explanation:

a. Correct – Movement of irradiated fuel bundles qualifies as a core alteration. Technical Specification Surveillance Requirement 3.3.1.2.2 requires an operable SRM is both the core quadrant where a core alteration is taking place and an adjacent quadrant. Surveillance Requirement 3.3.1.2.4 requires an SRM to have ≥3.0 cps with signal to noise ratio ≥2:1 or be within limits of Figure 3.3.1.2-1 to be considered operable. Both SRMs B and D are inoperable based on this requirement. With SRMs B and D inoperable, no core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant.

b. Incorrect – Two SRMs are operable (A and C), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant.

c. Incorrect – Two SRMs are operable (A and C), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant.

d. Incorrect – Two SRMs are operable (A and C), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant.

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LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s):	Technical Specification 3.3.1.2	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.2 without bases

Question Source:	Bank #		
	Modified Bank #	Vision SYSID 33954	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	
	55.43	6

Comments:

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water	Tier #		2
A2.03 - Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low CCW temperature	Group #		1
	K/A #	400000	
	Importance Rating	3.0	

Proposed Question: **# 87**

Unit 1 is operating at 100% power with the following:

- Degraded Service Water flow to RBCCW heat exchangers is resulting in high RBCCW temperatures.
- ON-RBCCW-101, Loss of Reactor Building Closed Cooling Water, is being executed.
- RBCCW supply header temperature is 105°F, up slow.
- Recirc pump A motor bearing temperatures are 186°F, up slow.
- Recirc pump B motor bearing temperatures are 180°F, up slow.
- Recirc pump A seal cavity temperatures are 189°F, up slow.
- Recirc pump B seal cavity temperatures are 185°F, up slow.

Which one of the following describes the required direction to be given, in accordance with ON-RBCCW-101?

Direct...

A. reducing Reactor power to below the 60% rod line. Tripping Recirc pump(s) and scramming the Reactor is NOT currently required.

B. reducing Reactor power to below the 60% rod line and tripping Recirc pump A, only. A Reactor scram is NOT currently required.

C. a Reactor scram and tripping of Recirc pump A, only.

D. a Reactor scram and tripping of both Recirc pumps A and B.

Proposed Answer:	**A**

Explanation:

a. Correct – ON-RBCCW-101 requires monitoring of Recirc pump motor bearing and seal cavity temperatures. If motor bearing temperatures approach 195°F or seal cavity temperatures approach 200°F, then Reactor power must be reduced to below the 60% rod line. Only if temperature reach these limits is further actions required (tripping pump(s) or scramming the Reactor). Since the given temperatures are below these limits but approaching, a Reactor power reduction is currently required only.

b. Incorrect – Recirc pump A temperatures are higher than those of Recirc pump B and approaching limits. This requires a Reactor power reduction. However, until Recirc pump A temperatures reach limits (motor bearing temp of 195°F or seal cavity temp of 200°F), a pump trip is not required.

c. Incorrect – Temperatures are approaching limits that will require a Reactor scram and tripping of pump(s), but have not yet reached the limits.

d. Incorrect – Temperatures are approaching limits that will require a Reactor scram and tripping of pump(s), but have not yet reached the limits.

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s):	AR-123-E05, ON-RBCCW-101	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

239002 SRVs

2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Level	RO	SRO
Tier #		2
Group #		1
K/A #	239002	
Importance Rating	4.6	

Proposed Question: **# 88**

Unit 1 is operating at 100% power with the following:

- A document review from the last outage shows the actual SRV relief and safety setpoints are set as follows:

Valve (PSV-141-xx)	Relief Setpoint	Safety Setpoint
F013A	1,135 psig	1,167 psig
F013B	1,127 psig	1,210 psig
F013C	1,144 psig	1,190 psig
F013D	1,139 psig	1,177 psig
F013E	1,145 psig	1,196 psig
F013F	1,147 psig	1,207 psig
F013G (ADS)	1,106 psig	1,252 psig
F013H	1,128 psig	1,194 psig
F013J (ADS)	1,118 psig	1,192 psig
F013K (ADS)	1,103 psig	1,248 psig
F013L (ADS)	1,114 psig	1,197 psig
F013M (ADS)	1,115 psig	1,245 psig
F013N (ADS)	1,116 psig	1,196 psig
F013P	1,127 psig	1,203 psig
F013R	1,138 psig	1,209 psig
F013S	1,125 psig	1,215 psig

Which one of the following describes the impact, if any, of these settings on Technical Specifications (TS)?

TS condition entry is...

A. NOT required based on these settings.

B. required for TS 3.4.3, S/RVs, only.

C. required for TS 3.5.1, ECCS – Operating, only.

D. required for TS 3.4.3, S/RVs, and TS 3.5.1, ECCS – Operating.

Proposed Answer:	B

Explanation:

a. Incorrect – SRVs G, K, and M safety setpoints are too high to be considered operable per TS SR 3.4.3.1. This leaves only 13 of the 16 SRVs operable for the safety function. TS 3.4.3 requires 14 SRVs to be operable for the safety function. Therefore TS 3.4.3 Condition A.1 entry is required.

b. Correct – SRVs G, K, and M safety setpoints are too high (>1241 psig) to be considered operable per TS SR 3.4.3.1. This leaves only 13 of the 16 SRVs operable for the safety function. TS 3.4.3 requires 14 SRVs to be operable for the safety function. Therefore TS 3.4.3 Condition A.1 entry is required.

c. Incorrect – SRVs G, K, and M are ADS valves. They are inoperable for their safety function, however this does not also make them inoperable for their ADS function. TS 3.5.1 deals with only the ADS function of SRVs, and therefore condition entry is not required in this case.

d. Incorrect – SRVs G, K, and M are ADS valves. They are inoperable for their safety function, however this does not also make them inoperable for their ADS function. TS 3.5.1 deals with only the ADS function of SRVs, and therefore condition entry is not required in this case.

Technical Reference(s): TS 3.4.3, TS 3.5.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TS 3.4.3 and TS 3.5.1 w/o bases

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

218000 ADS

2.2.12 - Equipment Control: Knowledge of surveillance procedures.

Level	RO	SRO
Tier #		2
Group #		1
K/A #	218000	
Importance Rating	4.1	

Proposed Question: **# 89**

Unit 1 is operating at 100% power with the following:

- It is discovered that Technical Specification 3.5.1, ECCS – Operating, Surveillance Requirement 3.5.1.11 was last satisfied 36 months ago:

SR 3.5.1.11	------------------------------NOTE------------------------------ Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months

- The surveillance CANNOT be immediately completed.
- A team is attempting to resolve issues to allow completion of the surveillance.
- Management has decided that an adequate Technical Specification risk evaluation CANNOT be performed for this issue.

Which one of the following describes the required administrative control of the associated LCO, in accordance with Technical Specifications?

A. Complete the surveillance within a maximum of 24 **MONTHS** from the time of discovery or then the associated LCO must be declared NOT met.

B. Complete the surveillance within a maximum of 6 **MONTHS** from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 24 **HOURS** from the time of discovery or then the associated LCO must be declared NOT met.

D. The associated LCO must be declared NOT met at this time.

Proposed Answer: **C**

Explanation:

a. Incorrect – TS SR 3.0.3 would allow a 24 month delay if a risk assessment is performed and the risk impact managed. The question states that an adequate risk evaluation cannot be performed, therefore the 24 month delay is not allowed, and the 24 hour delay time is limiting.

b. Incorrect – TS SR 3.0.3 only allows 24 hours in this case. 6 months is based on the 25% grace period allowed by TS SR 3.0.2.

c. Correct – TS SR 3.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 24 months), whichever is greater. However, the longer delay time is only allowed if a risk assessment is performed and the risk impact managed. The question states that an adequate risk evaluation cannot be performed, therefore the 24 month delay is not allowed, and the 24 hour delay time is limiting.

d. Incorrect – TS SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met.

Technical Reference(s): Technical Specification Surveillance Requirement 3.0.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC26R NRC #93 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
215005 APRM / LPRM	Tier #		2
A2.06 - Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions Recirculation flow channels upscale	Group #		1
	K/A #	215005	
	Importance Rating	3.5	

Proposed Question: **# 90**

Unit 1 is operating at 50% power during a power ascension with the following:

- APRM 3 has failed downscale and is now bypassed.
- Then, the Total Recirc Flow signal to APRM 1 fails upscale.

Which one of the following describes:

(1) the plant response to this failure, and

(2) the need for a condition entry in Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

A. (1) Rod block, but NO APRM vote.
(2) TS 3.3.1.1 condition entry is required.

B. (1) Rod block, but NO APRM vote.
(2) TS 3.3.1.1 condition entry is NOT required.

C. (1) Rod block and single APRM vote.
(2) TS 3.3.1.1 condition entry is required.

D. (1) Rod block and single APRM vote.
(2) TS 3.3.1.1 condition entry is NOT required.

Proposed Answer:	**A**

Explanation:

a. Correct – The flow signal failing upscale causes an APRM flow reference off-normal signal, which results in annunciator AR-103-E06 and a rod block. The upscale failure causes the flow biased scram setpoint for APRM 1 to be raised higher than normal while neutron flux remains constant, so no APRM vote is received. This non-conservative flow biased scram setpoint makes APRM 1 inoperable for TS Table 3.3.1.1-1 function 2.b, which requires 3 operable channels in the current Mode 1. APRM 3 is also inoperable for this function since it is bypassed. This leaves only two APRMs operable, which is less than the requirement. Therefore TS 3.3.1.1 Condition A must be entered.

b. Incorrect – TS 3.3.1.1 Condition A must be entered because only APRMs 2 and 4 remain operable for the flow-biased scram function, while three APRMs are required to be operable in Mode 1. No condition entry would be required if APRM 3 were operable.

c. Incorrect – The upscale failure causes the flow biased scram setpoint for APRM 1 to be raised higher than normal while neutron flux remains constant, so no APRM vote is received.

d. Incorrect – The upscale failure causes the flow biased scram setpoint for APRM 1 to be raised higher than normal while neutron flux remains constant, so no APRM vote is received. TS 3.3.1.1 Condition A must be entered because only APRMs 2 and 4 remain operable for the flow-biased scram function, while three APRMs are required to be operable in Mode 1. No condition entry would be required if APRM 3 were operable.

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): AR-103-E06, TS 3.3.1.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC24 Cert #15 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

201001 CRD Hydraulic

A2.08 - Ability to (a) predict the impacts of the following on the CRD HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of abnormal conditions or operations: Inadequate system flow

Level	RO	SRO
Tier #		2
Group #		2
K/A #	201001	
Importance Rating	2.8	

Proposed Question: **# 91**

Unit 1 is operating at 100% power when the in-service CRD Flow Control Valve fails closed.

Which one of the following describes a consequence of this malfunction and the required procedure response to be directed?

This malfunction results in ___(1)___.

Direct placing the standby CRD Flow Control Valve in service using ___(2)___.

A. (1) the control rod scram function being non-functional
   (2) ON-CRD-101, Control Rod Malfunction

B. (1) the control rod scram function being non-functional
   (2) AOP-155-001, Control Rod System Abnormal Operating Procedure

C. (1) control rod movement with RMCS being un-available
   (2) ON-CRD-101, Control Rod Malfunction

D. (1) control rod movement with RMCS being un-available
   (2) AOP-155-001, Control Rod System Abnormal Operating Procedure

Proposed Answer:	C

Explanation:

a. Incorrect – The CRD FCV is located downstream of the charging header. When the CRD FCV closes, it raises pressure in the charging header. Therefore, the control rod scram function is maintained.

b. Incorrect – The CRD FCV is located downstream of the charging header. When the CRD FCV closes, it raises pressure in the charging header. Therefore, the control rod scram function is maintained. Both ON-CRD-101 and AOP-155-001 contain guidance related to CRD FCV failures. ON-CRD-101 contains the specific guidance for placing the standby CRD FCV in service, which is required in this situation.

c. Correct – The CRD FCV is located downstream of the charging header and upstream of the drive and cooling headers. When the CRD FCV closes, it lowers pressure in the drive and cooling headers, while raising pressure in the charging header. Low pressure in the drive header results in control rod movement with RMCS being un-available. Both ON-CRD-101 and AOP-155-001 contain guidance related to CRD FCV failures. ON-CRD-101 contains the specific guidance for placing the standby CRD FCV in service, which is required in this situation.

d. Incorrect – Both ON-CRD-101 and AOP-155-001 contain guidance related to CRD FCV failures. ON-CRD-101 contains the specific guidance for placing the standby CRD FCV in service, which is required in this situation.

Technical Reference(s): ON-CRD-001, AOP-155-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:
Bank #
Modified Bank # NMP1 2013 NRC #92 (Note changes or attach parent)
New

Question History: Last NRC Exam

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
230000 RHR/LPCI: Torus/Pool Spray Mode	Tier #		2
2.2.40 - Equipment Control: Ability to apply technical specifications for a system.	Group #		2
	K/A #	230000	
	Importance Rating	4.7	

Proposed Question: **# 92**

Unit 1 is operating at 100% power with the following:

- Annunciator AR-109-B09, RHR LOOP A OUT OF SERVICE, alarms.
- Investigation reveals that the alarm was caused by trip of the breaker for HV-151-F028A, SUPP CHMBR SPR TEST SHUTOFF.
- The breaker appears damaged and CANNOT be reset.

Given the following Technical Specifications:

(1) TS 3.5.1 - ECCS - Operating

(2) TS 3.6.2.3 - Residual Heat Removal (RHR) Suppression Pool Cooling

(3) TS 3.6.2.4 - Residual Heat Removal (RHR) Suppression Pool Spray

Which one of the following identifies which of these Technical Specifications require condition entry due to this failure, if any?

A. None

B. (3), only

C. (2) and (3), only

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

Proposed Answer:	C

Explanation:

a. Incorrect – TS 3.6.2.3 and 3.6.2.4 require both of the two Suppression Pool Cooling and Spray subsystems operable. The failure of HV-151-F028A makes one of the two Suppression Pool Cooling and Spray subsystems inoperable. Therefore, both TS 3.6.2.3 and 3.6.2.4 condition entry is required.

b. Incorrect – Failure of HV-151-F028A closed also makes a subsystem of Suppression Pool Cooling inoperable, requiring condition entry for TS 3.6.2.3 also. Failure of HV-151-F027A would result in condition entry for TS 3.6.2.4 but not 3.6.2.3.

c. Correct – HV-151-F028A is normally closed and is required to be opened to establish either Suppression Pool Cooling or Suppression Pool Spray with RHR loop A. With the breaker tripped, HV-151-F028A cannot be opened, which makes one subsystem of Suppression Pool Cooling and one subsystem of Suppression Pool Spray inoperable. Only one other subsystem is available for each of these functions. TS 3.6.2.3 and 3.6.2.4 require two subsystems operable for each of these functions. Therefore both TS 3.6.2.3 and 3.6.2.4 condition entry is required. HV-151-F028A does not affect the ability of RHR loop A to perform its LPCI function, therefore TS 3.5.1 condition entry is not required.

d. Incorrect – HV-151-F028A does not affect the ability of RHR loop A to perform its LPCI function, therefore TS 3.5.1 condition entry is not required. Failure of both RHR pumps in loop A would result in condition entry for all three of the given TS.

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Technical Reference(s):	AR-109-B09, OP-149-004, TS 3.5.1, TS 3.6.2.3, TS 3.6.2.4	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	x	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

290002 Reactor Vessel Internals

2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls.

Level	RO	SRO
Tier #		2
Group #		2
K/A #	290002	
Importance Rating	4.0	

Proposed Question: **# 93**

Unit 1 is in a refueling outage with the following:

- Core shuffle phase II is in progress.
- A four cell section of the core is displayed on the following page.

Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with applicable fuel handling procedures?

A. The fuel is loaded correctly. Core shuffle may continue with no additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to startup.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Refuel Floor SRO.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Maintain the current configuration pending further Reactor Engineering direction.

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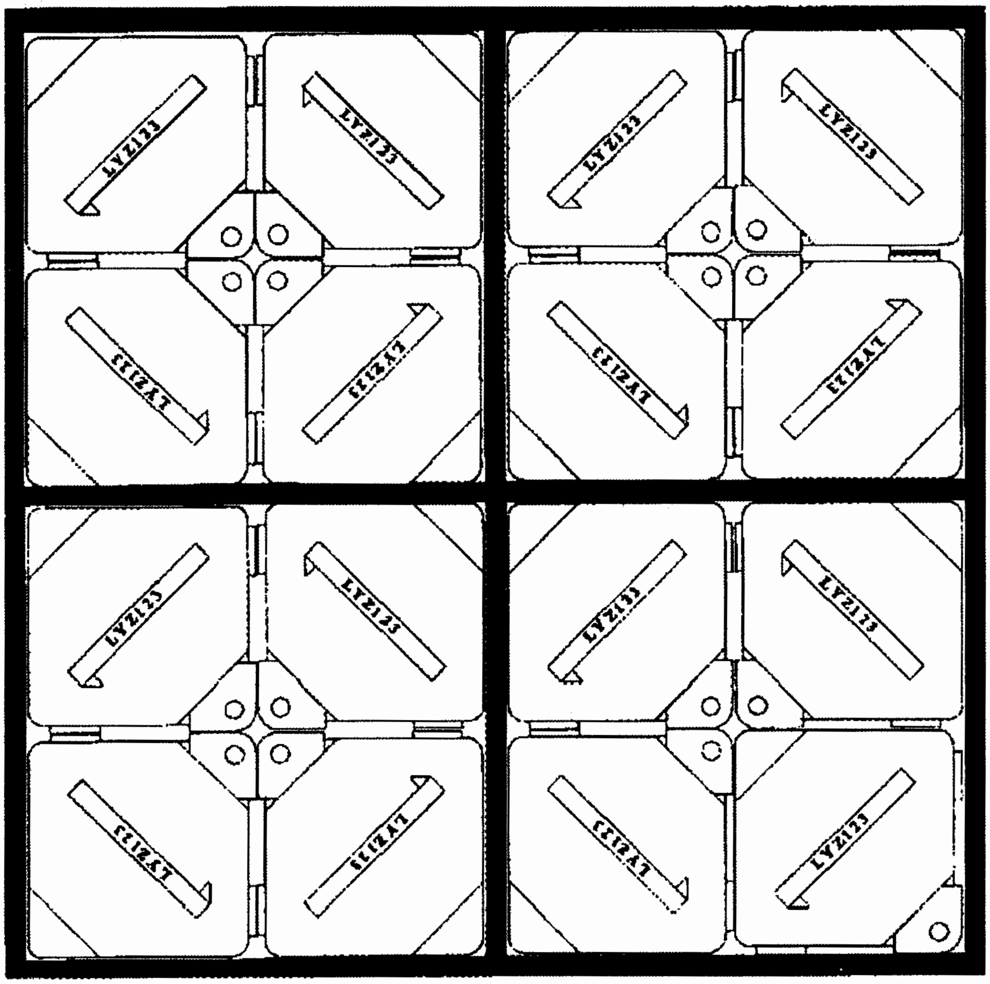

Proposed Answer: **D**

Explanation:

a. Incorrect – The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell and the channel fastener position.

b. Incorrect – This discrepancy requires entry into ON-FUEL-001, which requires stopping further fuel movement and entry into AOP-081-001. AOP-081-001 requires maintaining the fuel in the current configuration pending further Reactor Engineering direction.

c. Incorrect – This discrepancy requires entry into ON-FUEL-001, which requires stopping further fuel movement and entry into AOP-081-001. AOP-081-001 requires maintaining the fuel in the current configuration pending further Reactor Engineering direction.

d. Correct – The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell and the channel fastener position. This discrepancy requires entry into ON-FUEL-001, which requires stopping further fuel movement and entry into AOP-081-001. AOP-081-001 requires maintaining the fuel in the current configuration pending further Reactor Engineering direction.

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): OP-0RF-008 Attachments H and J, ON-FUEL-001, AOP-081-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # NMP1 2013 NRC #80

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 7

Comments:

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:	Level	RO	SRO
Conduct of Operations	Tier #		2
2.1.13 Knowledge of facility requirements for controlling vital / controlled access	Group #		2
	K/A #	G 2.1.13	
	Importance Rating	3.2	

Proposed Question: **# 94**

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

- A credible imminent insider threat has resulted in activation of the "Two Person Rule" in accordance with EP-PS-100, Emergency Director, Control Room - Emergency Plan Position Specific Instruction.
- The Security investigation is ongoing and the "Two Person Rule" will NOT be terminated for 2-3 hours.
- The Unit Supervisor determines that access to the Unit 1 Turbine Building is needed for an Operator to perform an inspection.
- There are NO other Operators currently available to accompany this Operator.
- The Operator requests to perform the inspection alone.

Which one of the following describes the correct response in accordance with EP-PS-100?

The Operator may...

A. NOT perform the inspection alone, but may be accompanied by a Security Officer.

B. NOT perform the inspection alone and must wait until another Operator is available.

C. perform the inspection alone because Operations is exempt from the "Two Person Rule".

D. perform the inspection alone because it is outside the area controlled by the "Two Person Rule".

Proposed Answer:	**A**

Explanation:

a. Correct – The "Two Person Rule" requires everyone in a vital area (which includes the Turbine Building) to be accompanied by another person. The individuals must remain within line-of-sight of each other. It is desired, but not required, for individuals to have similar qualifications. Therefore, the Operator must be accompanied, but may be accompanied by a Security Officer, even though they have different qualifications.

b. Incorrect – It is desired, but not required, for individuals to have similar qualifications. Therefore, the Operator must be accompanied, but may be accompanied by a Security Officer, even though they have different qualifications.

c. Incorrect – The "Two Person Rule" requires **everyone** in a vital area (which includes the Turbine Building) to be accompanied by another person. Operations personnel are not exempted from this requirement.

d. Incorrect – The "Two Person Rule" requires everyone in a vital area (which includes the Turbine Building) to be accompanied by another person.

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LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): EP-PS-100 Tab G Task 14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Question Source: Bank # NMP2 2012 NRC #94

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam NMP2 2012 NRC #94

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:

Equipment Control

2.2.7 Knowledge of the process for conducting special or infrequent tests

Level	RO	SRO
Tier #		2
Group #		2
K/A #	G 2.2.7	
Importance Rating	3.6	

Proposed Question: **# 95**

A Special, Infrequent Or Complex Test/Evolution (SICT/E) is being performed, in accordance with NDAP-QA-0320, Special, Infrequent Or Complex Test/Evolution.

If the SICT/E takes longer than one shift, which one of the following identifies the position required to conduct the subsequent brief for new personnel involved with the SICT/E, in accordance with NDAP-QA-0320?

A. Shift Manager

B. Unit Supervisor

C. Test / Evolution Coordinator

D. Responsible Group Supervisor

Proposed Answer:	C

Explanation:

a. Incorrect – The Shift Manager would be expected to attend the brief, however NDAP-QA-0320 specifically requires subsequent briefings for new personnel to be conducted, at a minimum, by the Test / Evolution Coordinator.

b. Incorrect – The Unit Supervisor would be expected to attend the brief, however NDAP-QA-0320 specifically requires subsequent briefings for new personnel to be conducted, at a minimum, by the Test / Evolution Coordinator.

c. Correct – NDAP-QA-0320 requires subsequent briefings for new personnel to be conducted, at a minimum, by the Test / Evolution Coordinator.

d. Incorrect – The Responsible Group Supervisor would be expected to attend the brief, however NDAP-QA-0320 specifically requires subsequent briefings for new personnel to be conducted, at a minimum, by the Test / Evolution Coordinator.

SUSQUEHANNA STEAM ELECTRIC STATION
LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): NDAP-QA-0320 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # LOC24 Cert #98

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

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Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Radiation Control	Tier #		2
2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Group #		2
	K/A #	G 2.3.5	
	Importance Rating	2.9	

Proposed Question: **# 96**

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

- A radioactive liquid release is in progress from the Evaporator Distillate Sample Tank.
- Annunciators AR-107(207)-F06, RADWASTE EFFLUENT MON DNSCALE/INOP, alarms.
- An investigation finds that RITS-06433, LIQUID RADWASTE RADIATION, is malfunctioning and can NOT be repaired this shift.

Which one of the following describes the status of the release and the ability to recommence the release this shift, in accordance with ON-069-001, Abnormal Radiation Release Liquid?

A. The release has automatically terminated.

The release may recommence at half of the original release rate under the original release permit.

B. The release must be manually terminated.

The release may recommence at half of the original release rate under the original release permit.

C. The release has automatically terminated.

The release may NOT recommence under the original release permit.

The release may recommence with a new release permit and with Plant Effluent Radiation Monitor Inoperable requirements satisfied.

D. The release must be manually terminated.

The release may NOT recommence under the original release permit.

The release may recommence with a new release permit and with Plant Effluent Radiation Monitor Inoperable requirements satisfied.

Proposed Answer:	C

Explanation:

a. Incorrect – There is not a redundant radiation monitor installed for RITS-06433. With RITS-06433 inoperable, a new release permit is required with Plant Effluent Radiation Monitor Inoperable requirements satisfied prior to recommencing the release.

b. Incorrect – A downscale or inop failure of RITS-06433 causes an automatic termination of the release. There is not a redundant radiation monitor installed for RITS-06433. With RITS-06433 inoperable, a new release permit is required with Plant Effluent Radiation Monitor Inoperable requirements satisfied prior to recommencing the release.

c. Correct – A downscale or inop failure of RITS-06433 causes an automatic termination of the release. This also requires entry into ON-069-001. With RITS-06433 failed, ON-069-001 requires initiating a new release permit with Plant Effluent Radiation Monitor Inoperable requirements satisfied prior to recommencing the release.

d. Incorrect – A downscale or inop failure of RITS-06433 causes an automatic termination of the release.

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LOC27 NRC INITIAL LICENSE EXAMINATION
SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): ON-069-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:
Bank # LOC25 NRC #96
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam LOC25 NRC #96

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:

Emergency Procedures/Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Level	RO	SRO
Tier #		2
Group #		2
K/A #	G 2.4.4	
Importance Rating	4.7	

Proposed Question: **# 97**

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

- Movement of irradiated fuel was in progress in the Unit 1 Spent Fuel Pool.
- A fuel bundle was dropped.
- A valid Refuel Floor High Exhaust hi-hi radiation alarm is received.
- Refuel Floor High Exhaust radiation is 25 mR/hr, steady.
- A valid Standby Gas Treatment noble gas hi-hi release rate alarm exists.
- Standby Gas Treatment noble gas release rate is 3.2 E 8 μCi/min, steady.
- These conditions persist for 15 minutes.

Which one of the following describes the entry requirement for EO-000-105, Radioactivity Release Control?

Entry into EO-000-105 is...

A. required based on the Refuel Floor High Exhaust radiation level.

B. required based on the Standby Gas Treatment noble gas release rate.

C. NOT required because the radiation/release conditions remain below the required thresholds.

D. NOT required because the radiation/release conditions are due to a non-primary system discharge.

Proposed Answer:	**B**

Explanation:

a. Incorrect – The Refuel Floor High Exhaust alarm condition directly requires entry into EO-000-104, but not EO-000-105.

b. Correct – EO-000-105 entry is based on "Offside Rad Release Rate Above Alert Anticipated". One condition that satisfies this criterion is any noble gas release rate >2.0 E 8 µCi/min for ≥15 minutes. Since SBGT noble gas release rate is greater than this threshold, EO-000-105 entry is required.

c. Incorrect – SBGT noble gas release rate is greater than the threshold requiring EO-000-105 entry (>2.0 E 8 µCi/min for ≥15 minutes).

d. Incorrect – While subsequent actions in EO-000-105 are dependent on whether the release is from a primary or non-primary system, entry into EO-000-105 is required even though the release is from a non-primary system.

Note: The question meets SRO level guidelines because it requires analysis of release rates against Emergency Action Level criteria, not just memorization of EOP entry conditions.

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Technical Reference(s): EO-000-105 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # LOC25 NRC #79 (Note changes or attach parent)

New

Question History: Last NRC Exam

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
Conduct of Operations	Tier #		2
2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, fuel depletion, etc.	Group #		2
	K/A #	G 2.1.43	
	Importance Rating	4.3	

Proposed Question: **# 98**

Unit 1 is operating at 30% power with the following:

- Reactor Recirc pump (RRP) 1A spuriously tripped earlier in the shift.
- Maintenance has fixed the issue that caused the spurious trip.
- Preparations are underway to restart RRP 1A per OP-164-001, Reactor Recirculation System.
- RRP 1A suction and discharge bypass valves have been opened.
- Rod line is 45%.
- The ΔT between Recirc loops is 70^{o}F, steady.

Which one of the following describes the acceptability of these conditions for the start of RRP 1A, in accordance with OP-164-001 and Technical Specifications?

A. Conditions are acceptable for the start of RRP 1A.

B. The rod line is acceptable for the start of RRP 1A, but the ΔT between Recirc loops is NOT acceptable.

C. The ΔT between Recirc loops is acceptable for the start of RRP 1A, but the rod line is NOT acceptable.

D. Both the rod line and the ΔT between Recirc loops are NOT acceptable for the start of RRP 1A.

Proposed Answer:	B

Explanation:

a. Incorrect – The ΔT between Recirc loops is NOT acceptable (>50°F) per OP-164-001 and TS SR 3.4.10.4.

b. Correct – The rod line is acceptable (≤60%) per OP-164-001. The ΔT between Recirc loops is NOT acceptable (>50°F) per OP-164-001 and TS SR 3.4.10.4. Starting RRP 1A with this high of a ΔT between Recirc loops is not allowed due to the potential for a more severe reactivity addition than analyzed in FSAR section 15.4.4.

c. Incorrect – The rod line is acceptable (≤60%) per OP-164-001. The ΔT between Recirc loops is NOT acceptable (>50°F) per OP-164-001 and TS SR 3.4.10.4.

d. Incorrect – The rod line is acceptable (≤60%) per OP-164-001.

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Technical Reference(s):	OP-164-001 Section 2.4, TS 3.4.10	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

SUSQUEHANNA STEAM ELECTRIC STATION
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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Examination Outline Cross-reference:

Emergency Procedures/Plan

2.4.26 Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

Level	RO	SRO
Tier #		2
Group #		2
K/A #	G 2.4.26	
Importance Rating	3.6	

Proposed Question: **# 99**

Both Units are operating at 100% power when the Motor-Driven Fire pump (OP512) becomes unavailable due to a breaker failure.

Which one of the following describes the required condition entry, if any, for TRM 3.7.3.1, Fire Suppression Water Supply System?

TRM 3.7.3.1 condition entry is...

A. NOT required because both the Diesel Engine Driven Fire Pump (0P511) and the Backup Diesel Engine-Driven Fire Pump (0P592) remain operable.

B. required. The Motor-Driven Fire pump (OP512) must be restored to operable status within 7 days, only.

C. required. The Motor-Driven Fire pump (OP512) must be restored to operable status within 7 days OR an alternate fire suppression water supply must be established within 7 days.

D. required. The Motor-Driven Fire pump (OP512) must be restored to operable status within 7 days AND an alternate fire suppression water supply must be established within 24 hours.

Proposed Answer:	C

Explanation:

a. Incorrect – While the Backup Diesel Engine-Driven Fire Pump (0P592) may be cross-tied to the main fire protection system loop to satisfy the Required Action of TRM 3.7.3.1, it does not satisfy the TRO such that condition entry is avoided.

b. Incorrect – While restoring the Motor-Driven Fire pump (OP512) within 7 days would satisfy TRM 3.7.3.1 Require Action A.1, it is not required as long as an alternate fire suppression water supply is established within 7 days.

c. Correct – TRM 3.7.3.1 Condition A must be entered because one of the two required fire suppression water supply subsystems is inoperable with the Motor-Driven Fire pump (OP512) inoperable. While the Backup Diesel Engine-Driven Fire Pump (0P592) may be cross-tied to the main fire protection system loop to satisfy the Required Action of TRM 3.7.3.1, it does not satisfy the TRO such that condition entry is avoided. The Required Actions allow either restoration of the Motor-Driven Fire pump (OP512) within 7 days or establishing an alternate fire suppression water supply within 7 days.

d. Incorrect – The Required Actions allow either restoration of the Motor-Driven Fire pump (OP512) within 7 days or establishing an alternate fire suppression water supply within 7 days. These more restrictive Required Actions would be correct for a loss of two fire suppression water supply subsystems.

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Technical Reference(s): TRM 3.7.3.1 and bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TRM 3.7.3.1 w/o bases

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge x

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

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Comments:

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Examination Outline Cross-reference:

Equipment Control

2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Level	RO	SRO
Tier #		2
Group #		2
K/A #	G 2.2.18	
Importance Rating	3.9	

Proposed Question: **# 100**

Unit 1 is in a refueling outage and Operations With Potential For Draining Reactor Vessel (OPDRVs) are in progress in accordance with NDAP-QA-0326, Operations With Potential For Draining Reactor Vessel.

Which one of the following describes the required review frequency of NDAP-QA-0326 Attachment C, OPDRV Log, and the individuals required to complete this review, in accordance with NDAP-QA-0326?

	Required Review Frequency	Individuals Responsible
A.	Once each 12 hours	OCC Risk Assessors
B.	Once each 12 hours	Senior Reactor Operators
C.	Once each 72 hours	OCC Risk Assessors
D.	Once each 72 hours	Senior Reactor Operators

Proposed Answer:	B

Explanation:

a. Incorrect – The review of NDAP-QA-0326 Attachment C must be performed by two SROs. OCC Risk Assessors are responsible for other outage risk assessment tasks, but not this specific review.

b. Correct – The review of NDAP-QA-0326 Attachment C must be performed each 12 hour shift by two SROs.

c. Incorrect – The review of NDAP-QA-0326 Attachment C must be performed each 12 hour shift by two SROs. OCC Risk Assessors are responsible for other outage risk assessment tasks, but not this specific review.

d. Incorrect – The review of NDAP-QA-0326 Attachment C must be performed each 12 hour shift.

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SENIOR REACTOR OPERATOR WRITTEN EXAMINATION

Technical Reference(s): NDAP-QA-0326 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 3

Comments: