	WRITTEN/ORAL EXAMINATION COVERSHEET

Trainee Name:			
Employee Number:		**Site:**	DAEC
Examination Number/Title: *Series A, Rev. 0, 2009 NRC Reactor Operator Initial License Exam*			
Training Program: *Initial License Training*			
Course/Lesson Plan Number(s): *50007 / Various*			
Total Points Possible: 75	**PASS CRITERIA:** ≥ **80%**		**Grade: __/**75 **= __%**
Graded by:			**Date:**
Co-graded by (if necessary):			**Date:**

EXAMINATION RULES

1.	References may not be used during this examination, unless otherwise stated.
2.	Read each question carefully before answering. If you have any questions or need clarification during the examination, contact the examination proctor.
3.	Conversation with other trainees during the examination is prohibited.
4.	Partial credit will not be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5.	Rest room trips are limited and only one examinee at a time may leave.
6.	For exams with time limits, you have **360 (6 Hours)** minutes to complete the examination.
7.	Feedback on this exam may be documented on QF-1040-13, Exam Feedback Form. Contact Instructor to obtain a copy of the form.

EXAMINATION INTEGRITY STATEMENT

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

"I acknowledge that I am aware of the Examination Rules stated above. Further, I have not given, received, or observed any aid or information regarding this examination prior to or during its administration that could compromise this examination."

Examinee's Signature: **Date:**

REVIEW ACKNOWLEDGEMENT

"I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the examination questions with the instructor to ensure my understanding.

Examinee's Signature: **Date:**

1 Point

1. Following a normal plant shutdown, a Shutdown Cooling (SDC) system startup has commenced IAW OI 149, RHR System.

   Which one of the following describes a constraint associated with the SDC startup?

   a. The RHR loop temperature should be warmed to 100°F to minimize thermal shock to the Recirc-RHR loop intertie.

   b. Both Recirc Pump Discharge Valves are normally left open to keep the Recirc loop warm.

   c. The Recirc Pump Discharge BYPASS Valves are normally left open to keep the Recirc loop warm and need not be closed.

   d. RHR Loop “A” is the preferred loop for the Shutdown Cooling mode because MO-2010 RHR CROSSTIE allows starting and stopping Shutdown Cooling without having to enter the Torus Area.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K1.03	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause- effect relationships between SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and the following: Recirculation loop temperature

Proposed Question: RO Question # 1

Proposed Answer: C

A: Incorrect – This would be a misconception of the relationship. The RHR loop temperature must be warmed to 150 degrees to minimize shock

B: Incorrect - Per the OI, during SDC startup, the Recirc Pump discharge valves in both Recirc loops shall remain closed to prevent SDC flow from bypassing the core.

C: Correct – Per OI 149 Note at step 5.5 (48) (a) - The Recirc Pump discharge valves in both Recirc loops shall remain closed to prevent SDC flow from bypassing the core. The Recirc Pump discharge bypass valves are normally left open to keep the Recirc loop warm and need not be closed. A Recirc Pump suction valve may be closed instead of the respective pump's discharge valve if Reactor coolant temperature is <212°F.

D: Incorrect – "B" is the preferred loop IAW OI-149 Section 5.5 1 NOTE.

Technical Reference(s): OI 149 rev 110 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:		
	55.41	3, 10
	55.43	

(3) Mechanical components and design features of the reactor primary system.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

2. The plant is operating at near full power with all systems in a normal lineup.

   The equalizing valve on the level transmitter selected for input to the feedwater level control system vibrates open.

   With no operator action, what will be the response of the plant and why?

   a. Reactor water level will rise to 211 inches causing a Main Turbine trip and reactor scram. This is due to the level transmitter inputting a low level signal to the feedwater level control system causing the FRVs to open and increase flow.

   b. Reactor water level will increase but stabilize at less than 211 inches due to feed flow/steam flow mismatch.

   c. Reactor Vessel level will decrease and the reactor will scram at 170 inches. This is due to the level transmitter inputting a high level signal to the feedwater level control system causing the FRVs to close and decrease flow.

   d. Reactor water level will decrease but stabilize at greater than 170 inches due to feed flow/steam flow mismatch.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	259002	K1.01	
	Importance Rating	3.8		

Knowledge of the physical connections and/or cause- effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: RPS

Proposed Question: RO Question # 2

Proposed Answer: C

A: Incorrect - The equalizing valve opening will cause the level input to the FW level control system to sense a high level. This would cause actual level to lower.

B: Incorrect - The equalizing valve opening will cause the level input to the FW level control system to sense a rising high level. This would cause level to lower.

C: Correct – SD 644, page 14,Opening the equalizing valve on the transmitter, a DP cell, would cause the transmitter to sense a high level and provide that input to the FRVs thru the Master level Controller summer circuit. Level would continue to lower until the RPS scram setpoint was reached (170 inches)

D: Incorrect - With the system continuing to sense a high level, vessel level would continue to lower and would not stabilize

Technical Reference(s): GFES Comp Chap 7, pg 20
SD 644, Rev.9, page 39 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	

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

1 Point

3. Which one of the following describes the power supply arrangement for the ADS Logic?

   a. ADS Logic A normal power supply is backed up by ADS Logic B normal power supply.

   b. ADS Logic B normal power supply is backed up by ADS Logic A normal power supply.

   c. ADS Logic A normal power supply is backed up by LLS Logic B backup power supply.

   d. ADS Logic B normal power supply is backed up by LLS Logic A backup power supply.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	218000	K2.01	
	Importance Rating	3.1		

Knowledge of electrical power supplies to the following: ADS logic

Proposed Question: RO Question # 3

Proposed Answer: B

A: Incorrect - ADS SRV Logic A has NO backup power supply

B: Correct – SD 183.1, page 22 - Normal and backup 125 VDC power for the ADS logic circuits and operation of the Safety/Relief Valves is provided from the two plant 125 VDC battery systems. 125 VDC battery 1D1 normally supplies power for ADS logic “A” and for all Safety/Relief Valves except LLS valve PSV-4407. 125 VDC battery 1D2 normally supplies power for ADS logic “B” and for LLS valve PSV-4407. Except for ADS logic “A”, loss of the normal 125 VDC power supply will deenergize a relay and automatically shift to the other 125 VDC supply as a backup. ADS logic “A” does not have a backup 125 VDC supply.

C: Incorrect - ADS SRV Logic A has NO backup power supply

D: Incorrect - ADS SRV Logic B normal power supply is backed up by LLS SRV Logic A normal power supply (same as ADS SRV Logic A normal power supply)

Technical Reference(s): SD-183.1 Rev 7. page 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam: **2007 NRC**

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

4. Which one of the following power supply loss or losses would cause BOTH an APRM "C" INOP trip and a loss of the APRM "C" trip indication on Panel 1C05?

   A loss of power from ______.

   a. 120 VAC from RPS Bus “A".

   b. 120 VAC from RPS Bus “A" and 120 VAC Instrument AC Control Power.

   c. 120 VAC from RPS Bus “A" and 120 VAC Uninterruptible AC Control Power.

   d. 120 VAC Instrument AC Control Power and 120 VAC Uninterruptible AC Control Power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	215005	K2.02	
	Importance Rating	2.6		

Knowledge of electrical power supplies to the following: APRM channels

Proposed Question: RO Question # 4

Proposed Answer: B

A: Incorrect - 120 VAC Instrument AC Control Power provides trip indication at the panel

B: Correct - 120 VAC from RPS Bus “A” supplies APRM Channels A, C, and E, LPRM Group B and Flow Units A and C. 120 VAC power from the Instrument AC Control Power system is used for LPRM meter lamps; for the four-rod display; trip, bypass, and inoperative indication and two IRM/APRM recorders at 1C-05.

C: Incorrect - 120 VAC Uninterruptible AC Control Power supplies recorder power for the 2 recorders not powered by Instrument AC

D: Incorrect - 120 VAC from RPS Bus “A" causes an APRM INOP trip.

Technical Reference(s): SD-878.3, Rev.11, page 41 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	

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

1 Point

5. The plant is at full power with all equipment in a normal lineup.

A partial loss of GSW occurs. No operator actions have been taken.

Which one of the following describes:

(1) loads DIRECTLY cooled by GSW,

AND

(2) conditions under which AOP 411-GSW Abnormal Operation, requires a Fast Power Reduction

a. (1) Recirc MG set lube oil
(2) Both Recirc MG Lube oil temperatures reach 210°F.

b. (1) Drywell Coolers
(2) Drywell Cooler outlet temperature is rising faster than GSW inlet temperature is rising.

c. (1) Main Generator Hydrogen Coolers
(2) Main Generator Hydrogen Cooler temperature is rising and cannot be maintained.

d. (1) Recirc Pump Motor Coolers
(2) Recirc Pump Motor Cooler outlet temperatures reach 250°F.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	400000	K3.01	
	Importance Rating	2.9		

Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Proposed Question: RO Question # 5

Proposed Answer: C

A: Incorrect - This would require a reactor scram
B: Incorrect - GSW does not provide drywell cooling
C: Correct - Per AOP 411 page 22, - Follow up step 11
D: Incorrect - GSW does not directly provide cooling to the Recirc Pump Motor Coolers

Technical Reference(s): AOP 411, rev 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	4, 7, 10
	55.43	

(4) Secondary coolant and auxiliary systems that affect the facility.
(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

6. The plant is operating at 100% when annunciator 1C03A(C-7) SRV BELLOWS FAILURE is activated. At 1C21, the Bellows Integrity light is OFF for PSV-4405 ADS Relief Valve.

   Which one of the following statements describes how PSV-4405 functions to control reactor pressure with this condition?

   a. PSV-4405 will NOT open if ADS is initiated but will open on its safety setpoint of 1140 psig.

   b. PSV-4405 will NOT open on its safety setpoint of 1140 psig but will open if ADS is initiated.

   c. PSV-4405 will NOT open on either its safety setpoint of 1140 psig or if ADS is initiated.

   d. PSV-4405 will open on its safety setpoint of 1140 psig or if ADS is initiated.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	239002	K3.01	
	Importance Rating	3.9		

Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following: Reactor pressure control

Proposed Question: RO Question # 6

Proposed Answer: B

A: Incorrect - The ADS function will still be operable

B: Correct - Per ARP 1C03A(C-7), Bellows failure does not necessarily mean the valve is leaking; however, it does make the safety (mechanical) actuation portion of the relief valve inoperative.

C: Incorrect - The ADS function will still be operable

D: Incorrect - The safety relief function will not operate

Technical Reference(s): ARP 1C03A(C-7) rev 48 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	

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

1 Point

7. The plant is at 100% power. All equipment is operable.

At T = 0 minutes – A grid disturbance occurs resulting in the diesel generator start logics sensing an Essential Bus undervoltage signal of <65% of rated bus voltage.

At T = 2 minutes – All Offsite Power is lost.

At T = 4 minutes - Drywell Pressure reaches 2.0 psig.

Which one of the following describes when the Diesel Generators start and the sequence of the RHR and Core Spray pump starts?

a. The Diesel Generators will start at T=0
The Core Spray pumps sequence on first

b. The Diesel Generators will start at T=0
The RHR pumps sequence on first

c. The Diesel Generators will start at T=2
The Core Spray pumps sequence on first

d. The Diesel Generators will start at T=2
The RHR pumps sequence on first

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	264000	K4.05	
	Importance Rating	3.2		

Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: Load shedding and sequencing

Proposed Question: RO Question # 7

Proposed Answer: A

A: Correct - Per SD 149 page 21, SD 151 page 20, If a LOCA signal exists, the RHR and Core Spray pumps will automatically sequence onto the 1A3[1A4] once power is restored. The Core spray pumps sequemnce on first. Per SD 324, page 33 - DGS will auto start on Loss of Offsite Power (LOOP) signals of <65% of rated voltage on the secondaries of both the Startup and Standby transformers or degraded essential bus voltage of <92.5% of rated voltage for 8-8.5 seconds.

B: Incorrect - The RHR and Core Spray pumps will sequence on and not start immediately. The Core Spray pumps sequence on first

C: Incorrect - The DGs will on start on the essential bus undervoltage at T=0, They will already be running when and loaded on the bus when all Offsite Power is lost.

D: Incorrect - The DGs will on start on the essential bus undervoltage at T=0, They will already be running when and loaded on the bus when all Offsite Power is lost. The RHR and Core Spray pumps will sequence on and not start immediately. The Core Spray pumps sequence on first

Technical Reference(s): SD 324 Rev 10, SD 149 Rev 11, SD 150 Rev 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

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

1 Point

8. With the Reactor Mode Switch in REFUEL, which one of the following conditions will result in Source Range Monitors producing a Reactor Scram?

   a. ONE SRM Channel indicating >5 x 10^5 counts per second with the Shorting Links removed.

   b. ONE SRM Channel indicating >5 x 10^5 counts per second with the Shorting Links installed.

   c. At least 2 SRM Channels indicating >1 x 10^5 counts per second with the Shorting Links removed.

   d. At least 2 SRM Channels indicating >1 x 10^5 counts per second with the Shorting Links installed.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	215004	K4.02	
	Importance Rating	3.4		

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals

Proposed Question: RO Question # 8

Proposed Answer: A

A: Correct - The SRM upscale trip occurs if any channel counts exceed 5×10^5 cps. A light is energized on the SRM drawer and signals are sent to a 1C05 indicator. The upscale trip unit also generates a scram signal for use by the Reactor Protective System. This scram signal is only functional during initial fuel loading and low power physics testing. At other times, this scram is removed by the installed shorting links.

B: Incorrect - the shorting links must be removed

C: Incorrect - 1×10^5 Is the upscale alarm and not the scram setpoint

D: Incorrect - the shorting links must be removed, 1×10^5 Is the upscale alarm and not the scram setpoint

Technical Reference(s): SD 878.1 rev 6 page 18; SD 358 rev 7 page 13 & 14 (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	

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

9. HPCI is being started for the quarterly full flow test surveillance. HPCI has reached the 2000 gpm flow rate when the ramp generator fails to its low limit.

   Which one of the following describes the response of the HPCI System?

   a. HPCI speed and flow lower.

   b. HPCI trips due to a loss of reference signal.

   c. HPCI will be unaffected while in automatic ONLY.

   d. HPCI remains at the same speed and flow.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	206000	K5.06	
	Importance Rating	2.6		

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : Turbine speed measurement: BWR-2,3,4

Proposed Question: RO Question # 9

Proposed Answer: A

A: Correct - Per SD 152, page 10 - The outputs of the flow controller and ramp generator are applied to a low value signal selector, which passes the lower of the two signals. Because the ramp generator output is less than the flow controller output, the HPCI turbine speed will be controlled solely by the ramp generator during startup. Throughout the entire startup transient, the flow controller output calls for speed consistent with the flow controller setting until such time that pump flow reaches the setpoint of the flow controller and the ramp function signal exceeds the flow controller signal.

B: **Incorrect -** The reference signal is not lost. The signal now going through the LVG will cause speed and flow to decrease

C: **Incorrect** - The ramp generator and the flow error signals feed into a LVG. The lower of the two signals is passed to the turbine control valve. If the ramp generator fails low, this low signal will pass to the TCV and it will close causing decreased speed and flow whether in manual or automatic.

D: **Incorrect** - Both speed and flow will decrease.

Technical Reference(s): SD 152 Rev 10 page 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Question Source:	Bank #	X – HC 2007	
	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	5, 7
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

10. Complete the following statements:

    The IRM detectors operate in the ___(1)___ region of the gas amplification curve. In addition, a decrease in IRM detector argon gas pressure will cause the IRM detectors to be ___(2)___ sensitive.

    a. (1) proportional
       (2) more

    b. (1) proportional
       (2) less

    c. (1) ionization
       (2) more

    d. (1) ionization
       (2) less

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K5.01	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detector operation

Proposed Question: RO Question # 10

Proposed Answer: D

A: Incorrect: Proportional region is incorrect. In addition, argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive. Plausible: The Fuel Loading Chambers used with SRMs for initial loading operated in the proportional region, but are no longer used. In addition, argon could be confused for a quench gas, and if it did act as a quench gas, then the detector would be more sensitive with less quench gas.

B: Incorrect: Proportional region is incorrect. Plausible: The Fuel Loading Chambers used with SRMs for initial loading operated in the proportional region, but are no longer used. In addition, the second half of the answer is correct.

C: Incorrect: Argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive. Plausible: Ionization region is correct. In addition, argon could be confused for a quench gas, and if it did act as a quench gas, then the detector would be more sensitive with less quench gas.

D: Correct: IRM detectors operate in the ionization region. Argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive.

Technical Reference(s): SD 878.1, Rev 6, page 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: IRM 78.1.1.4 (As available)

Question Source:	Bank #	X – Pilgrim 2007	
	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	2, 5, 7
	55.43	

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

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

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

1 Point

11. The plant is in normal full power operation with no equipment out of service.

    Which of the following describes the plant response if all 250 VDC were to be lost?

    a. Both Instrument AC buses would transfer their respective Regulating Transformers and the plant would remain stable.

    b. Both Instrument AC buses would be lost and the plant would scram due to loss of all Instrument AC power.

    c. The Uninterruptible AC bus would transfer to the Regulating Transformer and the plant would remain stable.

    d. The Uninterruptible AC bus would be lost and the plant would scram due to loss of all Uninterruptible AC power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262002	K6.02	
	Importance Rating	2.8		

Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : D.C. electrical power

Proposed Question: RO Question # 11

Proposed Answer: C

A: Incorrect - Instrument AC would be unaffected. The Inst AC Regulating transformers are supplied by 480 VAC.

B: Incorrect - Instrument AC would be unaffected. The Inst AC Regulating transformers are supplied by 480 VAC.

C: Correct - The Uninterruptable AC regulating transformers would supply the bus thru the static switch upon loss of all 250 VDC (see SD 357 Figure page 5)

D: Incorrect - Uninterruptible AC power would not be lost . It would be supplied via 480 VAC thru Uninterruptable AC Regulating transformer.

Technical Reference(s): SD -357, Rev 7, Figure 1
AOP 388, Rev 18, pg 3 (AutoAct) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X - DAEC	
	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	

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

1 Point

12. You are the At-The-Controls operator.

    A plant event occurred resulting in the need to use Standby Liquid Control (SBLC) as an injection source.

    A loss of Instrument AC 1Y11 occurs before the operator attempts to place SBLC in service.

    How will the SBLC system and indications respond?

    a. Both pumps start. Squib valve continuity lights extinguish. Pump discharge pressure and flow will indicate zero.

    b. Both pumps start. Squib valve continuity lights will be illuminated. Pump discharge pressure and flow will indicate normally.

    c. Both pumps start. Squib valve continuity lights extinguish. Pump discharge pressure and flow will indicate normally.

    d. Neither pump will start and the squib valves loss of continuity lights will be illuminated. Pump discharge pressure and flow will indicate zero.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	211000	K6.03	
	Importance Rating	3.2		

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM : A.C. power

Proposed Question: RO Question # 12

Proposed Answer: A

A: Correct – Per SD 153, page 26 - The Instrument AC System provides power for several of the SBLC instruments: SBLC Storage Tank Level (LI-2600A), SBLC Pump Discharge Pressure (PI-2605), SBLC System Flow (FI-2620), Injection Valve Position (V26-0032). Therefore, flow indication will fail to zero. The pumps & squib valves have not lost power (powered from 1B34 and 1B44)

B: Incorrect – The pumps & squib valves have not lost power (powered from 1B34 and 1B44) and the Squib valve indication will be extinguished. Loss of 1Y11 will cause flow and pressure to indicate zero.

C: Incorrect – Loss of 1Y11 will cause flow and pressure to indicate zero.

D: Incorrect – Both pumps will start, the squib valves will fire.

Technical Reference(s): SD 153, Rev 7, page 26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

13. RCIC is in operation taking a suction from the suppression pool and discharging to the reactor vessel. The RCIC flow controller is in automatic and is set at 400 gpm. Indicated flow is 400 gpm

Which one of the following describes the system response, steady state to steady state, if the RCIC MIN FLOW BYPASS Valve - MO-2510, fails open?

a. Flow to the reactor will decrease. Total system flow will stabilize at 400 gpm.
Indicated flow will remain at 400 gpm.

b. Flow to the reactor will remain constant. Total system flow will stabilize at greater than 400 gpm.
Indicated flow will remain at 400 gpm.

c. Flow to the reactor will increase. Total system flow will stabilize at 400 gpm.
Indicated flow will be greater than 400 gpm.

d. Flow to the reactor will remain constant. Total system flow will stabilize at greater than 400 gpm.
Indicated flow will be greater than 400 gpm.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	217000	A1.01	
	Importance Rating	3.7		

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC flow

Proposed Question: RO Question # 13

Proposed Answer: B

A: Incorrect - flow to the reactor will remain constant and system flow will be > 400 gpm.

B: Correct - The flow element for the RCIC flow controller is downstream of the min flow valve and the min flow discharges to the suppression pool. Therefore, flow to the reactor will remain at the flow controller setpoint (400 gpm) irrespective of min flow line status. When the min flow isolation valve is opened, total system flow will increase due to the new flowpath introduced to the system that is not seen by the flow control valve or the indication.

C: Incorrect – flow to the reactor will remain constant and system flow will be >400 gpm.

D: Incorrect – Indicated flow will remain at 400 gpm.

Technical Reference(s): SD 150, Rev 6, Figure 3 system diagram (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: RCIC 3.8.1.2 (As available)

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, 7
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

14. The Div 1 125 VDC battery charger is being operated in the equalize mode.

Which one of the following describes:

(1) the voltage relationship between the charger and the batteries

AND

(2) the design rating of the batteries if a loss of AC power occurred?

a. (1) In equalize, the charger output to the battery will be a higher voltage than when in the float mode
   (2) The 125 VDC batteries are sized to supply emergency power for a 4-hour time period.

b. (1) In equalize, the charger output to the battery will be a lower voltage than when in the float mode
   (2) The 125 VDC batteries are sized to supply emergency power for a 4-hour time period.

c. (1) In equalize, the charger output to the battery will be a higher voltage than when in the float mode
   (2) The 125 VDC batteries are sized to supply emergency power for an 8-hour time period.

d. (1) In equalize, the charger output to the battery will be a lower voltage than when in the float mode
   (2) The 125 VDC batteries are sized to supply emergency power for an 8-hour time period.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 A1.01	
	Importance Rating	2.5	

Ability to predict and/or monitor changes associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Proposed Question: RO Question # 14

Proposed Answer: A

A: Correct – Per SD 375, pages 7 and 17. Each charger can be placed in the equalizing mode with a switch on the battery charger. When in the equalizing mode, the charger output will be a higher voltage than when in the float mode.
The Plant 125v DC Power Supply System consists of two 125 VDC batteries each provided with its own charger and sized to supply emergency power for a 4-hour time period.

B: Incorrect - in equalize the charger output to the battery will be a higher (not lower) voltage than when in the float mode

C: Incorrect – design is for 4 hours

D: Incorrect - design is for 4 hours and - in equalize the charger output to the battery will be a higher (not lower) voltage than when in the float mode

Technical Reference(s): SD 375, Rev 7, pages 7 and 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 5, 7
55.43

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

15. The plant is operating at 100% power. All systems are operable.

Instrument Air Dryer 1T-265A was in service when control power to the dryers was momentarily lost while it automatically switched between 1L150 and 1L21.

Which one of the following describes:

(1) the effect on instrument air header pressure

AND

(2) actions required, if any, IAW OI 518.1 “Instrument, Service and Breathing Air Systems” in regard to continued operation of the air dryer.

a. (1) Instrument air header pressure would lower and the Service Air Header would isolate.
   (2) Dispatch an operator to reset the dryer logic locally.

b. (1) Instrument air header pressure may fluctuate slightly.
   (2) Dispatch an operator to reset the dryer logic locally.

c. (1) Instrument air header pressure would lower and the Service Air Header would isolate.
   (2) No action is required.

d. (1) Instrument air header pressure may fluctuate slightly.
   (2) No action is required.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	300000	A2.01	
	Importance Rating	2.9		

Ability to predict the impacts of the following on the INSTRUMENT AIR SYSTEM: and use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions

Proposed Question: RO Question # 15

Proposed Answer: B

A: Incorrect- Service Air header isolates at 82 psig. In this situation the loss of power places both drying chambers in service and pressure should not significantly lower

B: Correct – Per SD 518, page 20 The air dryer would shut down when control power is lost. If control power to dryers will be momentarily lost while the control power automatically switches between 1L150 to 1L21 or vice versa, de-energize and then re-energize affected dryer by placing dryer power handswitch HS-3046A[B] in OFF for 5 seconds and then return to ON. This will reset dryer logic. The air dryer would shut down when control power is lost and header pressure would lower with no flowpath.
SD 518 page 20 - The Bypass Valve will automatically open on two conditions. If Instrument Air pressure downstream of the dryers falls to 85 psig, the dryer Bypass Valve CV-3026 automatically opens and control room annunciator 1C07B (B-10) INSTRUMENT AIR DRYERS 1T-265A/B LO DISCH PRESSURE is activated. If differential pressure across the dryer units reaches 15 psid, the dryer Bypass Valve CV-3026 automatically opens and control room annunciator 1C07B (C-10) INSTRUMENT AIR DRYERS 1T-265A/B HIΔP is activated.

C: Incorrect – Service Air header isolates at 82 psig. In this situation the loss of power places both drying chambers in service and pressure should not significantly lower. Action is required to reset the dryer logic.

D: Incorrect - Action is required to reset the dryer logic.

Technical Reference(s): SD 518, Rev 8, page 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5, 7
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

16. The plant was operating at 100% when a recirc line break occurred.

- Reactor pressure is at 410 psig and stable
- Drywell Pressure is at 3.4 psig and rising slowly
- Reactor level is at 60 inches and steady
- Core Spray pumps are running
- Core Spray Valves INBD INJECT MO-2117 and MO-2137 are closed
- Core Spray MIN FLOW BYPASS VALVES MO-2104 and MO-2124 are open

Which one of the following describes the response of the Core Spray System and actions required, if any, in regard to INBD INJECT VALVES and MIN FLOW BYPASS VALVES?

a. The Core Spray Inboard Injection Valves should have opened and must be manually opened.
The Core Spray Min Flow Bypass Valves will auto-close ONLY when the Injection Valves are fully open.

b. The Core Spray Inboard Injection Valves are closed and will open once reactor pressure lowers to below the shut off head of the Core Spray pumps.
The Core Spray Min Flow Bypass Valves will auto-close when Core Spray system flow reaches 600 gpm.

c. The Core Spray Inboard Injection Valves should have opened and must be manually opened.
The Core Spray Min Flow Bypass Valves will auto-close when Core Spray system flow reaches 600 gpm.

d. The Core Spray Inboard Injection Valves are closed and will open once reactor pressure lowers to below the shut off head of the Core Spray pumps.
The Core Spray Min Flow Bypass Valves will auto-close ONLY when the Injection Valves are fully open.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001	A2.08
	Importance Rating	3.1	

Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM: and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Valve openings

Proposed Question: RO Question # 16

Proposed Answer: C

A: Incorrect – The min flow bypass valve will close when system flow reaches 600 gpm

B: Incorrect - The injection valves should have opened at 450 psig RPV pressure and must be opened manually based on the step stating to“verify” they open.

C: Correct – OI 151, pages 6 and 7, steps 4.0 (2) and (3). When system flow reaches 600 gpm, as indicated on (A[B] CORE SPRAY PUMP) INJECT/TEST FLOW indicator FI-2110 [FI-2130] on Panel 1C03, verify MIN FLOW BYPASS MO-2104 [MO-2124] valve CLOSES.
When reactor vessel pressure drops below the low pressure permissive setpoint of 450 psig, verify that the INBD INJECT MO-2117 [MO-2137] valves OPEN to inject to the reactor vessel.
The injection valves should have opened at 450 psig RPV pressure and must be opened manually based on the step stating to“verify” they open.

D: Incorrect - The injection valves should have opened at 450 psig RPV pressure and must be opened manually based on the step stating to“verify” they open.
The min flow bypass vlv will close when system flow reaches 600 gpm.

Technical Reference(s): OI 151 Rev 57 steps 4.0 (2) and (3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5, 7
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

17. While operating at 75% power, the PB-5831A, "A" SBGT TEST push-button is depressed on SBGT control panel 1C24A.

    Which of the following would automatically occur as a result of the above action?

    a. The “A” SBGT would start, a secondary containment isolation would occur and normal Reactor Building Ventilation would isolate.

    b. The “A” SBGT would start, a secondary containment isolation would NOT occur and Reactor Building Ventilation would NOT isolate.

    c. The “A” SBGT would NOT start, a secondary containment isolation would occur and normal Reactor Building Ventilation would isolate.

    d. The “A” SBGT would NOT start, a secondary containment isolation would NOT occur and Reactor Building Ventilation would NOT isolate.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	261000	A3.03	
	Importance Rating	3.0		

Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including : Valve operation

Proposed Question: RO Question # 17

Proposed Answer: B

A: Incorrect – a secondary containment isolation would NOT occur, Reactor Building Ventilation would NOT isolate.

B: Correct – Per SD 170, page 16 - The SBGT trains can also be manually initiated without causing any isolation using SBGT TEST pushbuttons PB-5831A (B) with the mode switch in AUTO. Use of the SBGT TEST pushbutton will initiate the associated SBGT train, however, will not initiate secondary containment isolation.

C: Incorrect - With the SBGT Mode Select Switch in “MANUAL,” no actions occur when the test PB is depressed.

D: Incorrect - With the SBGT Mode Select Switch in “MANUAL,” no actions occur when the test PB is depressed.

Technical Reference(s): SD 170, Rev 10 page16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: SBGT 7.2.1.1 (As available)

Question Source:	Bank #	X - 19287	
	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	

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

1 Point

18. The plant is operating at full power. All systems are operable.

Then, the Fuel Pool Exhaust rad monitors alarm and both are reading 8.5 mr/hr.

Which one of the following describes PCIS Group(s) and valve(s) affected by this alarm?

a. PCIS Group 3. The Well Water Drywell Cooling Water Supply and Return Valves close causing a rise in drywell temperatures.

b. PCIS Groups 2 & 3. The drywell and torus sample supply valves isolate preventing any liquid or gaseous samples from being taken from the drywell or torus.

c. PCIS Group 3. The drywell and torus sample supply valves isolate preventing any liquid or gaseous samples from being taken from the drywell or torus.

d. PCIS Group 2. The RHR sample valves isolate which prevents sampling of torus water when in torus cooling or reactor water when in shutdown cooling.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	223002	A3.02	
	Importance Rating	3.5		

Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: Valve closures

Proposed Question: RO Question # 18

Proposed Answer: C

A: Incorrect – This alarm produces a Group 3 isolation only. The well water Drywell Cooling Water Supply and Return Valves are PCIS Group 7 valves

B: Incorrect – This alarm produces a Group 3 isolation only. The drywell and torus sample supply valves isolate are Group 3

C: Correct – Per SD 959-1, page 10 table 1 - Fuel Pool Exhaust High Radiation, 8 mr/hr or Inop is a Group 3 isolation.
Group 3 includes a large number of valves, dampers, and fans. As direct isolations, this group includes the Primary Containment Atmosphere Control Valves for Drywell and Torus Ventilation and Purge; the Containment Nitrogen Compressor Suction and Discharge Valves; the Jet Pump Sample, Liquid Return, and Sample Station Exhaust PASS Valves; and Recirculation Pump Seal Purge Valves.

D: Incorrect - This alarm produces a Group 3 isolation only

Technical Reference(s): SD 959-1 Rev 8, page 10 table 1 and page 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: PCIS 76.1.1.7 (As available)

Question Source:	Bank #		
	Modified Bank #	X - 20170	(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	

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

Bank Question 20170 - Correct Answer: C

Which of the following describes how the PASS responds to a valid PCIS Group II signal?

A. The jet pump sample valves isolate preventing any reactor samples from being taken.

B. The torus sample supply and return valves isolate preventing any liquid samples from being taken.

C. The RHR sample valves isolate which prevents sampling of torus water when in torus cooling or reactor water when in shutdown cooling.

D. The drywell and torus sample supply valves isolate preventing any liquid or gaseous samples from being taken off the drywell or torus.

1 Point

19. Which one of the following describes how the Scram Discharge Volume High Level trip may be bypassed?

    a. It must be bypassed by placing the keylocked High Water Level Bypass switch in BYPASS.
    The mode switch may be in either the SHUTDOWN or REFUEL position.

    b. It must be bypassed by placing the keylocked High Water Level Bypass switch in BYPASS.
    The mode switch must be in the SHUTDOWN position ONLY.

    c. It is automatically bypassed after a time delay.
    The mode switch may be in either the SHUTDOWN or REFUEL position.

    d. It is automatically bypassed after a time delay.
    The mode switch must be in the SHUTDOWN position ONLY.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	212000	A4.04	
	Importance Rating	3.9		

Ability to manually operate and/or monitor in the control room: Bypass SCRAM instrument volume high level SCRAM signal

Proposed Question: RO Question # 19

Proposed Answer: A

A: Correct - Per ARP 1C05 – E1, The CRD Scram Discharge Volume High Water Level Scram trip is bypassed using a keylocked High Water Level Bypass switch if the Reactor Mode switch is in SHUTDOWN or REFUEL.
B: Incorrect - The mode switch may be in SHUTDOWN or REFUEL.
C: Incorrect - The keylock switch must in BYPASS. No time delay applies
D: Incorrect – The keylock switch must in BYPASS. No time delay applies. The mode switch may be in SHUTDOWN or REFUEL.

Technical Reference(s): ARP 1C05B (E-1) Rev 81 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

20. The “A” SBDG is being paralleled to 1A3.

- Incoming voltage is slightly higher than running voltage
- The synchroscope is rotating slowly in the clockwise direction

The SBDG output breaker is closed when the synchroscope pointer reaches the 12 o'clock position.

Which one of the following will occur immediately after the breaker is closed?

a. The SBDG will supply only MWe to the grid.

b. The SBDG will supply both MWe and MVAR to the grid.

c. The SBDG output breaker will trip open on reverse power.

d. The SBDG will supply MWe to the grid but absorb MVAR from the grid.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262001	A4.04	
	Importance Rating	3.6		

Ability to manually operate and/or monitor in the control room: Synchronizing and paralleling of different A.C. supplies

Proposed Question: RO Question # 20

Proposed Answer: B

A: Incorrect – MVARs are also being supplied

B: Correct – OI 304.2 Rev 77 & Theory portion of electrical fundamentals - current flows from the DG to the grid. Both MWE and MVARs will be supplied

C: Incorrect – the output breaker will not trip open unless the difference between running (grid) and incoming (DG) exceeds specified values or the synchroscope is not close to the 12 o'clock position. This answer is correct if the synch scope is running the counterclockwise direction

D: Incorrect – the SBDG will supply, NOT absorb MVARs

Technical Reference(s): OI 304.2 Rev 77
GFES Comp Ch 5, pg 51 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	ID 18809	
	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

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

1 Point

21. The plant is operating at full power with all equipment operable. RHR Loop “A” is then placed in torus cooling in preparation for a surveillance test.

    Which ONE of the following Technical Specification LCO(s) is impacted by placing RHR in Torus Cooling?

    The LCO(s) for _____.

    a. Low Pressure Coolant Injection ONLY

    b. RHR Suppression Pool Spray ONLY

    c. RHR Suppression Pool Spray AND Low Pressure Coolant Injection

    d. RHR Suppression Pool Cooling AND Low Pressure Coolant Injection

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	203000	2.2.40	
	Importance Rating	3.4		

Equipment Control:: Ability to apply Technical Specifications for a system.

Proposed Question: RO Question # 21

Proposed Answer: A

A: Correct – Per OI 149 Continuous Recheck Statement at 5.4 – **IF** Torus Cooling is operating when LPCI is required to be Operable, **THEN** LPCI shall be declared inoperable and the Technical Specifications for ECCS-Operating and ECCS-Shutdown complied with.

B: Incorrect – PER OI 149 – LPCI is required to be declared inoperable

C: Incorrect - ONLY the LPCI LCO must be entered

D: Incorrect - ONLY the LPCI LCO must be entered

Technical Reference(s): OI 149 Rev 110, Section 5.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	WTS	
	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	

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

1 Point

22. The plant is at 30% power with a power ascension in progress. All equipment is operable.

Then, the following alarm annunciates:

- 1C05A (E-2) APRM FLOW UNIT UPSCALE, INOP OR COMPARE ERROR

Investigation determines the alarm is due to a COMPARATOR error.

Which one of the following describes:

(1) the plant response to the alarm

AND

(2) the back panel 1C37 indication(s)?

a. (1) ONLY a Rod Block has occurred.
   (2) At least 2 flow units would normally indicate a COMPARATOR alarm at back panel 1C37.

b. (1) A Rod Block and a Half Scram have occurred.
   (2) At least 2 flow units would normally indicate a COMPARATOR alarm at back panel 1C37.

c. (1) ONLY a Rod Block has occurred.
   (2) ONLY the flow unit with the flow mismatch would normally indicate a COMPARATOR alarm at back panel 1C37.

d. (1) A Rod Block and a Half Scram have occurred.
   (2) ONLY the flow unit with the flow mismatch would normally indicate a COMPARATOR alarm at back panel 1C37.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005	2.4.46
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions. (APRMs)

Proposed Question: RO Question # 22

Proposed Answer: A

A: Correct – Per 1C05A (E-2), A rod withdraw block occurs in all Modes of Operation. A flow signal mismatch which produces a compare error in one flow unit will normally produce a compare error in another (unbypassed) flow unit.

B: Incorrect – ONLY a Rod Block occurs

C: Incorrect - A flow signal mismatch which produces a compare error in one flow unit will normally produce a compare error in another (unbypassed) flow unit.

D: Incorrect – ONLY a Rod Block occurs . A flow signal mismatch which produces a compare error in one flow unit will normally produce a compare error in another (unbypassed) flow unit.

Technical Reference(s): 1C05A (E-2) Rev 64 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7, 10
55.43

(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

23. Which one of the following describes the relationship between HPCI and ventilation exhaust?

    a. The HPCI barometric condenser vacuum pump exhausts directly to the Offgas Stack, BOTH of the OG STACK EXH FANS, 1V-EF-18A & B, must be ON when HPCI is running.

    b. The HPCI barometric condenser vacuum pump exhausts directly to the Offgas Stack, At least one OG STACK EXH FAN, 1V-EF-18A[B], must be ON when HPCI is running.

    c. The HPCI barometric condenser vacuum pump exhausts through the SBGT trains, At least one OG STACK EXH FAN, 1V-EF-18A[B], must be ON when HPCI is running. At least one SBGT train has to be aligned for AUTO but does not have to be in operation.

    d. The HPCI barometric condenser vacuum pump exhausts through the SBGT trains, At least one OG STACK EXH FAN, 1V-EF-18A[B], must be ON and at least one SBGT train must be running when HPCI is running.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	261000	K1.06	
	Importance Rating	3.0		

Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: High pressure coolant injection system; Plant-specific

Proposed Question: RO Question # 23

Proposed Answer: C

A: Incorrect – HPCI vacuum pump exhaust is directed to the SBGT system and 1V-EF-18A[B]-fan must be running.

B: Incorrect - HPCI vacuum pump exhaust is directed to the SBGT system

C: Correct – Per SD 152, page 16 - A vacuum pump maintains system vacuum by transferring non-condensible gases to the Standby Gas Treatment System. Per STP 3.5.1-05 rev 45 Section 6.0 (prereqs), at least one SBGT system has to be aligned for AUTO and one OG STACK EXH FAN, 1V-EF-18A[B], must be on when HPCI is running.

D: Incorrect – SBGT is not required to be in operation if HPCI is running

Technical Reference(s): SD 152 Rev 10 page 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X - 20154	
	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	

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

1 Point

24. You have indications of an SRV leaking.

    IAW AOP 683 “Abnormal Safety Relief Valve Operation”, which one of the following tailpipe temperatures would first indicate that the SRV is OPEN?

    When tailpipe temperature reaches ______.

    a. 212 degrees F.

    b. 251 degrees F.

    c. 312 degrees F.

    d. 544 degrees F.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	218000	A1.01	
	Importance Rating	3.4		

Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures

Proposed Question: RO Question # 24

Proposed Answer: B

A: Incorrect – This would indicate a leaking SRV.

B: Correct – Per AOP 683 NOTE page 3, An SRV or SV is considered OPEN based on the following: At 1C21- Tailpipe temperature at TR-4400 above 250ºF.

C: Incorrect – This would be the approximate temperature at the tailpipe pressure but IAW the AOP the SRV is considered open at >250 degrees F.

D: Incorrect – This temperature is not achievable given tailpipe pressure. It is the coolant temperature at 1000 psig.

Technical Reference(s): AOP 683 Rev 8 (Attach if not previously provided)

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

Learning Objective: (As available)

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	5
	55.43	

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

25. The plant was operating at 80% power when a condensate pump tripped. The plant was scrammed and RCIC is being placed in service manually.

    As RCIC reaches rated flow, the following annunciators alarm:
    - 1C04C-A7 – RCIC “A” LOGIC MAN/AUTO ISOL INITIATED
    - 1C04C-A8 – RCIC “B” LOGIC MAN/AUTO ISOL INITIATED

    What is the response of the RCIC system to this alarm?

    a. RCIC Turbine Stop Valve MO-2405 will close.
    RCIC Turbine Steam Supply Valve MO-2404 remains open.
    RCIC Inboard Steam Line Isolation Valve MO-2400 will close.
    RCIC Outboard Steam Line Isolation Valve MO-2401 will close.

    b. RCIC Turbine Stop Valve MO-2405 will close.
    RCIC Turbine Steam Supply Valve MO-2404 will close.
    RCIC Inboard Steam Line Isolation Valve MO-2400 remains open.
    RCIC Outboard Steam Line Isolation Valve MO-2401 remains open.

    c. RCIC Turbine Stop Valve MO-2405 remains open.
    RCIC Turbine Steam Supply Valve MO-2404 remains open.
    RCIC Inboard Steam Line Isolation Valve MO-2400 will close.
    RCIC Outboard Steam Line Isolation Valve MO-2401 will close.

    d. RCIC Turbine Stop Valve MO-2405 remains open.
    RCIC Turbine Steam Supply Valve MO-2404 will close.
    RCIC Inboard Steam Line Isolation Valve MO-2400 remains open.
    RCIC Outboard Steam Line Isolation Valve MO-2401 remains open.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A4.03	
	Importance Rating	3.4	

(RCIC) Ability to manually operate and/or monitor in the control room: System valves

Proposed Question: RO Question # 25

Proposed Answer: A

A: Correct - Per SD 150, this is a RCIC trip condition which closes the valves listed except for the MO-2404 which remains open
B: Incorrect - MO-2404 remains open, the MO-2400 & 2401 close
C: Incorrect – MO-2405 closes
D: Incorrect – MO-2404 remains open, the MO-2400 & 2401 close, MO-2405 closes

Technical Reference(s): SD 150 Rev 6 pages 18 & 23
1C04C-B8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

26. The plant is shutdown and RHR Loop A is in Shutdown Cooling with the “A” pump running.

RPV water level lowers to 50 inches.

How do the RHR pumps automatically respond to the signal?

a. The “A” pump remains in Shutdown Cooling.
   All other pumps start and operate on min flow.

b. The “A” pump trips then restarts on min flow.
   All other pumps start and operate on min flow.

c. The “A” pump trips and does not restart.
   “C” pump attempts to start and immediately trips.
   “B” and “D” pumps auto start.

d. The “A” pump remains in Shutdown Cooling.
   “C” pump attempts to start and immediately trips.
   “B” and “D” pumps auto start.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	205000	A3.02	
	Importance Rating	3.2		

Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) including: Pump trips

Proposed Question: RO Question # 26

Proposed Answer: C

A: Incorrect – The "A" pump trips. All others start but C trips

B: Incorrect - The "A" pump trips and does not restart. All others start but C trips

C: Correct – Per SD 149, page 22 - In the event a LOCA occurs when the RHR System is in the shutdown cooling mode, the RHR System will not automatically realign itself for LPCI injection. Operator actions required to initiate the LPCI mode of RHR include resetting the Group 4 Isolation Seal-In, restoring torus suction flowpath to the RHR pumps, and manually restarting the RHR pumps that have tripped.
Additionally, the SDC suction valves close on the LPCI signal (PCIS Group 4). The "C" RHR Pump breaker will receive a start signal but immediately trip. The trip occurs due to no suction path present to prevent pump damage. This is NOT a start permissive, it s a pump trip (SD-149 page 12).

D: Incorrect – The "A" pump trips. The "C" RHR Pump breaker will receive a start signal but immediately trip. The trip occurs due to no suction path present to prevent pump damage. This is NOT a start permissive, it s a pump trip (SD-149 page 12).

Technical Reference(s): SD 149 Rev 11 pages 12 & 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

27. Which one of the following describes the relationships between the RWCU system and its associated component cooling systems?

    a. RBCCW supplies cooling water to the shell side of the RWCU Non-Regen HX. Well Water DIRECTLY cools the RWCU pump coolers ONLY under Cold Shutdown conditions.

    b. RBCCW supplies cooling water to the tube side of the RWCU Non-Regen HX. Well Water DIRECTLY cools the RWCU pump coolers ONLY under Cold Shutdown conditions.

    c. RBCCW supplies cooling water to the shell side of the RWCU Non-Regen HX. Well Water is NOT DIRECTLY used for any cooling medium in the RWCU system.

    d. RBCCW supplies cooling water to the tube side of the RWCU Non-Regen HX. Well Water is NOT DIRECTLY used for any cooling medium in the RWCU system.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	204000	K1.04	
	Importance Rating	2.9		

Knowledge of the physical connections and/or cause- effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: Component cooling water systems

Proposed Question: RO Question # 27

Proposed Answer: C

A: Incorrect – Per SD 414 page 12, Well water may be used only under cold shutdown conditions to cool RBCCW, not RWCU.

B: Incorrect – Per SD 261 - RBCCW supplies cooling water to the shell side of the HX. Well water may be used only under cold shutdown conditions to cool RBCCW, not RWCU.

C: Correct – Per SD 261 page 10 - Water from the Reactor Building Closed Cooling Water System is circulated through the shell sides of the Non-Regenerative Heat Exchangers. No other CCW system connects to RWCU.

D: Incorrect - RBCCW supplies cooling water to the shell side of the HX.

Technical Reference(s): SD 261 Rev 6, page 10; SD 414 Rev 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #		(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	

(4) Secondary coolant and auxiliary systems that affect the facility

1 Point

28. Given the following plant conditions:

- The plant is in Mode 5
- “A” RPS is in Alternate being supplied by 1B32
- “B” RHR is in Supplemental Fuel Pool Cooling with Shutdown Cooling in service
- “A” RHR is in Torus Cooling

Then, annunciator 1C08A A-5 “Bus 1A3 Lockout Trip” alarms and

SBDG 1G-31 auto starts and _____

a. loads Bus 1A3. Torus Cooling is lost.

b. does NOT load Bus 1A3. Supplemental Fuel Pool Cooling with Shutdown Cooling is lost.

c. loads Bus 1A3. Torus Cooling remains in service.

d. does NOT load Bus 1A3. Supplemental Fuel Pool Cooling with Shutdown Cooling remains in service.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	233000	K2.02	
	Importance Rating	2.8		

Knowledge of electrical power supplies to the following: RHR pumps

Proposed Question: RO Question # 28

Proposed Answer: B

A: Incorrect – With a bus lockout, the DG output breaker will not close.

B: Correct – IAW ARP 1C08A-A-5 this alarm causes a bus load shed, DG start however the output breaker will not close
IAW OI 149 Section 10.1 – Describes how to start Supplemental FPC with SDC in service.

C: Incorrect – With a bus lockout, the DG output breaker will not close. B RHR will be lost

D: Incorrect – SDC is lost due to the 1A3 loss causing a trip of A RPS and subsequent Group 4 partial isolation.

Technical Reference(s): ARP-1C08A (A-5) Rev 75 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	WTS	
	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	

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

1 Point

29. The plant is at 100% power near the end of cycle with all control rods fully withdrawn. At this point the SCRAM INLET VALVE fails OPEN for control rod 18-27.

    Which of the following describes the Effects on the following over the next five (5) minutes?

    (1) Effect on reactor power

    AND

    (2) Effect on Scram Discharge Volume (SDV)

    a. (1) Reactor power will remain at 100% power.
       (2) There will be NO flow into the SDV.

    b. (1) Reactor power will be LOWER, but the plant will continue to operate at power.
       (2) There will be NO flow into the SDV.

    c. (1) Reactor power will be LOWER, but the plant will continue to operate at power.
       (2) There will be flow into the SDV, but it will be within the capacity of the SDV drain.

    d. (1) Reactor power will remain at 100%.
       (2) There will be flow into the SDV, but it will be within the capacity of the SDV drain.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	201003	K3.01	
	Importance Rating	3.2		

Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Reactor Power

Proposed Question: RO Question # 29

Proposed Answer: B

A:	Incorrect – Power will lower
B:	Correct – Per SD 255, Figure on page 24 - Charging water header pressure acting on the CRD mechanism under piston area will cause the rod to insert and power to lower. Since the scram outlet valve is closed, no flow will go to the SDV
C:	Incorrect – There will be no flow to the SDV
D:	Incorrect – The rod will insert and there will be no flow to the SDV

Technical Reference(s): SD 255 Rev 8 page 24 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X - 19990	
	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	6, 7
	55.43	

(6) Design, components, and functions of reactivity control mechanisms and instrumentation.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

30. The plant is operating at 100% power. Both Control Building intake radiation monitors, RM-6101A&B, have tripped. The Standby Filter Unit (SFU) Lockout Relays have tripped and all automatic actions have occurred as expected.

    The control room operator has just placed the B SFU train in STANDBY following the auto initiation.

    Which one of the following describes how the B SFU train will function?

    The B SFU train will _____

    a. auto initiate if the A SFU train flow rate lowers to 800 scfm or less.

    b. NOT auto initiate. Manual manipulations must be performed to place the B SFU train in operation.

    c. auto initiate ONLY if the high radiation condition clears and then occurs again.

    d. auto initiate immediately if the B SFU Mode Select HS-7316B is taken back to the AUTO position.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	290003	K4.01	
	Importance Rating	3.1		

Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following: System initiations/reconfiguration: Plant-Specific

Proposed Question: RO Question # 30

Proposed Answer: A

A: Correct – Per SD 730, page 28 - After both trains initiate, approximately 1000 scfm TOTAL is flowing through the system, one train can be manually transferred to standby mode by turning the mode switch to MAN and then back to AUTO. In this condition, mode switch in AUTO, lockout relay tripped and flow >800 scfm, the heater & exhaust fans are off and intake & discharge valves AV-7301A(B) and AV-7318A(B) are shut. The standby train will auto initiate, start the fan, energize the heaters, and open supply and discharge valves, if system flow drops to 800 scfm or less.

B: Incorrect – It would still auto initiate on low flow

C: Incorrect – It would also auto initiate on low flow

D: Incorrect – If placed in manual then back in auto, a low flow condition is required to start the fan

Technical Reference(s): SD 730 Rev 9, page 28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X - 22687	
	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	

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

1 Point

31. OI 644 "Condensate and Feedwater Systems" directs that the Condensate/Feed system be filled and vented prior to the FIRST Condensate Pump start.

    Which of the following is the reason for this action?

    a. To prevent pump damage due to exceeding pump vibration limits during pump startup.

    b. To prevent Condensate Pump vortex limits from being exceeded and vapor binding of the pump.

    c. To reduce the risk of water hammer and the system damage that could result.

    d. To prevent pump run out conditions in the Condensate Pump which will cause winding degradation.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	259001	K5.02	
	Importance Rating	2.5		

Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : Water hammer

Proposed Question: RO Question # 31

Proposed Answer: C

A: Incorrect – Not the reason per the OI. Although vibration may rise slightly during startup, the concern per the OI is water hammer in the system.

B: Incorrect – Not the reason per the OI. This would be a concern with a saturated liquid

C: Correct – Per OI 644 NOTE in Section 3.2 after step (9) - To reduce the risk of water hammer

D: Incorrect – Not the reason per the OI. This condition would occur if pump resistance to flow were zero.

Technical Reference(s): OI 644 Rev 109 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X – DAEC NRC 2002	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2002

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

10 CFR Part 55 Content:	55.41	5
	55.43	

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

1 Point

32. A plant event has occurred. The following conditions exist.

- Reactor level is 110” and lowering
- Reactor pressure is 550 psig and lowering
- Drywell pressure is 9.0 psig and rising
- Containment sprays are in service

Which one of the following conditions would cause an isolation of Containment Sprays? (Assume no operator actions)

a. One Wide Range Yarway reference leg break.

b. Reactor water level has lowered to 64 inches.

c. One Fuel Zone instrument variable leg break.

d. Loss of 1Y23, Uninterruptible AC power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	226001	K6.08	
	Importance Rating	2.7		

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE : Nuclear boiler instrumentation

Proposed Question: RO Question # 32

Proposed Answer: C

A: Incorrect – The wide range level instruments do not span the 2/3 core coverage level.

B: Incorrect – This is the LPCI initiation level setpoint

C: Correct – Per SD 880 page 39 - Containment spray cooling initiation logic receives a permissive signal from each of two reactor vessel level instruments. These switch contacts actuate when reactor vessel level is above -39" to allow containment spray cooling to be manually initiated. A variable leg break would give a false low level signal. Therefore, the 2/3 core coverage permissive for containment spray would not be met. The logic is shown in Figure 11 of SD 149

D: Incorrect – Uninterruptible AC power provides indication and recorder power for the GEMAC level instruments and would not affect the 2/3 core coverage containment spray interlock

Technical Reference(s): SD 149 Rev 11 page 54
SD 880 Rev 13 page 39 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X – 19015	
	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	

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

1 Point

33. A reactor startup is in progress. Reactor power is at 35% and the operators have just finished bypassing rod 02-19 on the Rod Worth Minimizer. There are now eight rods bypassed.

The control room operator selects rod 02-19 and attempts to withdraw the control rod in accordance with the pull sheet.

How will the control rod respond to the withdraw signal and what is the reason for that response?

a. The control rod will NOT withdraw. Since rod 02-19 has been bypassed, it can only be inserted.

b. The control rod will withdraw. Since rod 02-19 has been bypassed, the RWM is incapable of enforcing a rod block.

c. The control rod will NOT withdraw. This is due to having the maximum number of rods bypassed in the RWM.

d. The control rod will withdraw, as long as the RWM-CC keylock mode switch is in OPER.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	201006	A1.02	
	Importance Rating	3.4		

Ability to predict and/or monitor changes in parameters associated with operating the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) controls including: Status of control rod movement blocks; P-Spec(Not-BWR6)

Proposed Question: RO Question # 33

Proposed Answer: A

A: Correct – Per SD 878.8, page 16 - Rods that are bypassed are only allowed to be inserted. If the selected rod is a bypassed rod, then a withdraw block is applied.
B: Incorrect – a withdraw block is enforced
C: Incorrect – It will not withdraw because a withdraw block is enforced when the rod is bypassed in the RWM
D: Incorrect - a withdraw block is enforced

Technical Reference(s): SD 878.8 Rev 7 page 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X -19377	
	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	5, 7
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

34. With the plant operating at full power, an unisolable leak in the Main Turbine lube oil system occurs.

    The operators trip the Main Turbine with the Emergency Trip PB and turbine lube oil is secured.

    Which one of the following describes the recommended method of disconnecting the Main Generator from the grid IAW OI 698 “Main Generator” and the sequence for securing H2 Seal Oil.

    a. Verify that GENERATOR OUTPUT H BREAKER (OCB 0220) and GENERATOR OUTPUT I BREAKER (OCB 4290) have OPENED, then verify that the GENERATOR EXCITER FIELD BREAKER has tripped.
       Reduce generator hydrogen gas pressure to approximately 10 to 15 PSIG until the Hydrogen Seal Oil System is secured.

    b. Verify that GENERATOR OUTPUT H BREAKER (OCB 0220) and GENERATOR OUTPUT I BREAKER (OCB 4290) have OPENED, then verify that the GENERATOR EXCITER FIELD BREAKER has tripped.
       Secure the Hydrogen Seal Oil System, then reduce generator hydrogen gas pressure to approximately 10 to 15 PSIG.

    c. Manually trip the GENERATOR EXCITER FIELD BREAKER, then verify that GENERATOR OUTPUT H BREAKER (OCB 0220) and GENERATOR OUTPUT I BREAKER (OCB 4290) have OPENED.
       Reduce generator hydrogen gas pressure to approximately 10 to 15 PSI until the Hydrogen Seal Oil System is secured.

    d. Manually trip the GENERATOR EXCITER FIELD BREAKER, then verify that GENERATOR OUTPUT H BREAKER (OCB 0220) and GENERATOR OUTPUT I BREAKER (OCB 4290) have OPENED.
       Secure the Hydrogen Seal Oil System, then reduce generator hydrogen gas pressure to approximately 10 to 15 PSI.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	245000	A2.01	
	Importance Rating	3.7		

Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine trip

Proposed Question: RO Question # 34

Proposed Answer: A

A: Correct – Per OI 698 - Section 5.2.2 (a) (Recommended method) trip the turbine with the TURBINE EMERGENCY TRIP pushbutton and verify GENERATOR OUTPUT H BREAKER (OCB 0220) and GENERATOR OUTPUT I BREAKER (OCB 4290) are OPEN.
Step 5.2.7 - Verify GENERATOR EXCITER FIELD BREAKER tripped and GENERATOR FIELD BREAKER BACKUP green indicating light ON at 1C08.

AOP 693 – Followup Action 4 - If turbine lube oil is secured, reduce generator hydrogen gas pressure to approximately 10 to 15 PSI until the Hydrogen Seal Oil System is secured.

B: Incorrect - H2 pressure must be lowered prior to securing the seal oil system.

C: Incorrect - the exciter field breaker trips automatically and is verified tripped after verifying the output breakers open.

D: Incorrect - the exciter field breaker trips automatically. and is verified tripped after verifying the output breakers open. H2 pressure must be lowered prior to securing the seal oil system.

Technical Reference(s): AOP 693 Rev 12
OI 698 Rev 70 page 26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5, 10
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

35. The plant is operating at full power with a TIP trace in progress. The system is being operated in the automatic mode. Status of the TIPs is as follows:

- Channel A: Position 2 selected and inserting
- Channel C: Position 2 selected and withdrawing (tracing)

At this point a transient occurs that results in RPV level lowering to 165”.

Which one of the following describes the response, if any, of the TIP drives to this event?

a. Operation continues with no changes since RPV level is >119.5”.

b. Channel “A” reverses and both drives withdraw from the core to the in-shield position. The ball valves automatically close when associated detectors are at the in shield position.

c. Channel "A" continues insertion to the top of core, then withdraws to the in shield position without tracing.
Channel "C" continues withdrawing to the in shield position.
The operator then closes both ball valves by placing the MAN VALVE CONTROL to CLOSE.

d. Channel "A" continues insertion to the top of core, then withdraws to the in shield position without tracing.
Channel "C" continues withdrawing to the in shield position.
The ball valves automatically close when associated detectors are at the in shield position.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	215001	A3.03	
	Importance Rating	2.5		

Ability to monitor automatic operations of the TRAVERSING IN-CORE PROBE including: Valve operation: Not-BWR1

Proposed Question: RO Question # 35

Proposed Answer: B

A: Incorrect – The TIP probes withdraw when RPV level reaches 170”
B: Correct – Per SD 878.6 page 29 - The TIP System response to a Group 2 Containment Isolation (170 inches) is to retract to the shield any detectors that are inserted in the Reactor core. Then close the TIP Ball Valves.
C: Incorrect – Both probes immediately withdraw and the ball valves auto close
D: Incorrect - Both probes immediately withdraw

Technical Reference(s): SD 878.6 Rev 6 page 29 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: RO-83.03.01.05-02 (As available)

Question Source:	Bank #	X - 20210	
	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	

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

1 Point

36. The plant is operating at 90% power.

The CRS directs you to place Torus Cooling in service IAW OI 149 “RHR System” in advance of a RCIC quarterly surveillance test.

Which one of the following describes:

(1) How is Torus temperature controlled during this evolution

AND

(2) How RHRSW flow would be affected if a valid LPCI initiation signal occurred?

a. (1) RHR flow through the tube side of the RHR Heat Exchanger is throttled.
   (2) RHRSW flow through the RHR Heat Exchanger would be automatically secured.

b. (1) RHR flow through the tube side of the RHR Heat Exchanger is throttled.
   (2) RHRSW flow through the RHR Heat Exchanger would continue unchanged.

c. (1) RHR flow through the shell side of the RHR Heat Exchanger is throttled.
   (2) RHRSW flow through the RHR Heat Exchanger would be automatically secured.

d. (1) RHR flow through the shell side of the RHR Heat Exchanger is throttled.
   (2) RHRSW flow through the RHR Heat Exchanger would continue unchanged.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	219000	A4.05	
	Importance Rating	3.4		

RHR/LPCI: Torus/Pool Cooling Mode - Ability to manually operate and/or monitor in the control room: Heat exchanger cooling flow

Proposed Question: RO Question # 36

Proposed Answer: C

A: Incorrect – RHR flow is thru or around the shell side of the RHR HX
On a LPCI signal RHRSW pumps trip

B: Incorrect – On a LPCI signal, RHR flow is throttled around the RHR HX

C: Correct - Per OI 149, Section 5.4 step 11 - Close MO-2030 [1940] A[B] HEAT EXCH BYPASS valve if required.
Per SD 149 - The LPCI initiation signal overrides all modes of the RHR System (except shutdown cooling). The intent is to direct maximum system effort toward restoring and maintaining the reactor vessel water level, i.e., all pumps are started, all non-LPCI modes secured, and motor-operated valves positioned to direct the maximum amount of flow into the reactor vessel.

D: Incorrect – LPCI signal would shut the Torus Cooling valves, which would secure flow through the RHR heat exchanger.

Technical Reference(s): OI 149 Rev 110 page 37
SD 149 Rev 11 page 20
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

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

1 Point

37. Which instruments are required to provide Post Accident Monitoring (PAM) indication in the control room?

(1) Drywell Pressure
(2) RPV Fuel Zone Level
(3) RPV Wide Range Level
(4) RPV Metal Temperature
(5) Reactor Building Vent Shaft Rad Monitors

ONLY______.

a. (1), (2) and (5)

b. (1), (3) and (4)

c. (2), (4) and (5)

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

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	216000	2.4.3	
	Importance Rating	3.7		

(Nuclear Boiler instrumentation) Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

Proposed Question: RO Question # 37

Proposed Answer: D

A: Incorrect – Reactor Building Vent Shaft Rad Monitor is not a PAM instrument
B: Incorrect - RPV Metal Temperature is not a PAM instrument
C: Incorrect – RPV Metal Temperature and Reactor Building Vent Shaft Rad Monitors are not PAM instruments
D: Correct –IAW TS Table 3.3.3.1.1

Technical Reference(s): TS 3.3.3.1. Table 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

38. The plant is operating at 90% power with a power ascension in progress. All systems are operable.

Then, annunciator 1C06B (B-1), CONDENSATE DEMIN INLET/OUTLET HI ΔP, alarms.

Which one of the following describes:

(1) an action required as a result of this alarm

AND

(2) what is the concern due to the alarm condition?

a. (1) Verify that MO-1708 CONDENSATE DEMIN BYPASS has automatically opened and that Demin inlet/outlet ΔP is lowering to <40 psid.
   (2) Resin breakthrough can occur resulting in decreasing condensate water conductivity.

b. (1) Verify that MO-1708 CONDENSATE DEMIN BYPASS has automatically opened and that Demin inlet/outlet ΔP is lowering to <40 psid.
   (2) Resin breakthrough can occur resulting in additional chlorides and sulfates in the condensate system.

c. (1) Throttle open MO-1708 CONDENSATE DEMIN BYPASS to maintain system dP <40 psid.
   (2) Resin breakthrough can occur resulting in decreasing condensate water conductivity.

d. (1) Throttle open MO-1708 CONDENSATE DEMIN BYPASS to maintain system dP <40 psid.
   (2) Resin breakthrough can occur resulting in additional chlorides and sulfates in the condensate system.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	256000	A1.08	
	Importance Rating	2.7		

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: System water quality

Proposed Question: RO Question # 38

Proposed Answer: D

A: Incorrect - MO-1708 must be manually opened, Conductivity levels would increase not decrease

B: Incorrect - MO-1708 must be manually opened

C: Incorrect - Conductivity levels would increase not decrease

D: Correct – Per ARP 1C06B (B-1) – Operator Action step 3.2 - Maintain system dP <40 psid by throttling open MO-1708 CONDENSATE DEMIN BYPASS with handswitch HS-1708 at 1C06 or reducing reactor power as necessary to clear the alarm.

Per AOP 639 - Exceeding the 40 psid total F/D system ΔP as indicated at PDI-1707 at

Panel 1C06 or PDI-1708 at Panel 1C80 or exceeding the 25 psid individual F/D bed ΔP limits as indicated by PDI-1727A[B, C, D E] at Panel 1C80 may cause septa damage and resin breakthrough.

Technical Reference(s): AOP 639 Rev 29 page 9
ARP 1C06B (B-1) Rev 45 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 5, 10
55.43

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

39. The plant is operating at full power. All systems are operable.

Then, annunciator 1C08A (A-9), 125 VDC SYSTEM 1 TROUBLE, activates.

As the BOP operator you recognize that 125 VDC bus 1D10 is de-energized.

Which one of the following describes the effect on the associated bus normal supply breaker?

a. 4KV breaker overcurrent and undervoltage protection is lost. The 4KV supply breaker has tripped and cannot be reclosed from the control room.

b. 4KV breaker overcurrent and undervoltage protection is lost. The 4KV supply breaker remains closed.

c. 4KV breaker undervoltage protection is lost; however, overcurrent protection remains available. The 4KV supply breaker has tripped and cannot be reclosed from the control room.

d. 4KV breaker undervoltage protection is lost; however, overcurrent protection remains available. The 4KV supply breaker remains closed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004	AK1.05
	Importance Rating	3.3	

AK1.05 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Loss of breaker protection

Proposed Question: RO Question # 39

Proposed Answer: B

A: Incorrect – The breakers remain closed without breaker protection

B: Correct – Per AOP 302.1 – Automatic actions list. Loss of 1A1/1A3 breaker control occurs. Additionally all breaker protection is lost on loss of DC

C: Incorrect – The breakers will not trip. Control power to trip the breakers was lost. Additionally, all breaker protection is lost.

D: Incorrect – All breaker protection is lost.

Technical Reference(s): AOP 302.1 Rev 48 page 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7, 8
55.43

(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(8) Components, capacity, and functions of emergency systems

1 Point

40. Which one of the following events would bring a running Residual Heat Removal System pump closer to cavitation while injecting during the Low Pressure Coolant Injection (LPCI) mode of operation?

    a. Broken tailpipe on an SRV in the Suppression Chamber airspace

    b. A stuck closed Suppression Chamber to Drywell vacuum breaker

    c. Opened Safety Relief Valve

    d. CST leaking into the Suppression Pool

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295026	EK1.01	
	Importance Rating	3.0		

Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : Pump NPSH

Proposed Question: RO Question # 40

Proposed Answer: C

A: Incorrect – no effect on NPSH
B: Incorrect – improves NPSH
C: Correct – warmer water will lower pump NPSH
D: Incorrect – cooler water will improve NPSH

Technical Reference(s): EOP Basis Document, NPSH Graphs (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	5
	55.43	

1 Point

41. The plant was operating at 90% power when the following occurred:

- 1P201B, B RECIRC PUMP has tripped.
- Power has stabilized at 58% thermal power.

Which of the following describes how this event affects the reactor fuel?

Fuel failures _______

a. are less likely due to decreased margin to pre-conditioning limits.

b. are more likely due to decreased production of Iodine and Cadmium.

c. are more likely due to a decrease in Average Planar Linear Heat Generation Rate.

d. are less likely due to increased margin in the linear heat generation rate.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295001	AK1.03	
	Importance Rating	3.6		

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Thermal limits

Proposed Question: RO Question # 41

Proposed Answer: D

A: Incorrect - pre conditions margins actually increase

B: Incorrect - Pellet-clad interactions (PCI) type of fuel failures are enhanced by the production of embrittling agents (from fission) of Iodine and cadmium. As power drops, these isotopes are produced less and more margin to this type of failure is gained - not less margin.

C: Incorrect - APLHGR is a concern during LOCA conditions

D: Correct - As power drops core wide from reduced recirculation flow, the amount of energy produced per linear foot of fuel drops. This provides more margin to the LHGR limit.

Technical Reference(s): GFES Thermo, Chap 8, pg 26
GFES Thermo, Chap 9, pg 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # WTS
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 (2)
55.43

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

1 Point

42. The plant is operating at 50% power.

Then, low EHC pressure causes a Main Turbine trip.

Which one of the following describes how the plant is affected?

a. The Turbine Stop Valves Fast Close
The Turbine Control Valves Fast Close
Both Reactor Recirc Pumps Trip

b. ONLY the Turbine Control Valves Fast Close
Both Reactor Recirc Pumps Trip

c. The Turbine Stop Valves Fast Close
The Turbine Control Valves Fast Close
Both Reactor Recirc Pumps remain running

d. ONLY the Turbine Control Valves Fast Close
Both Reactor Recirc Pumps remain running

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295005	AK2.03	
	Importance Rating	3.2		

Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Recirculation system

Proposed Question: RO Question # 42

Proposed Answer: A

A: Correct – Per SD 693.2a – EHC low pressure results in a fast closure of the Turbine Control Valves & Stop valves. This will result in a reactor scram due to Turbine Control Valve Fast Closure signal (800# RETS pressure).
Per SD 693.1, page 19 - Turbine Control Valve Fast Closure is used to produce a Reactor Scram and End Of Cycle Recirculation Pump Trip (EOC-RPT) signal

B: Incorrect – ONLY the Turbine Control Valves Fast Closure would occur if a power to load unbalance turbine trip occurred.

C: Incorrect – Turbine Control Valve Fast Closure is used to produce a Reactor Scram and End Of Cycle Recirculation Pump Trip (EOC-RPT) signal

D: Incorrect - EOC-RPT from the TCV fast closure causes both reactor recirc pumps to trip which would occur if the turbine trip was caused by the power to load unbalance circuitry.

Technical Reference(s): SD 693.1 Rev 9 page 19
SD 693.2a Rev 5 table A page 37 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 (7)
55.43

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

1 Point

43. The plant is at full power. All systems are operable.

A local alarm horn sounds at panel 1C179 “Cardox Fire Protection Control Panel” and control room annunciator 1C40 (F-6) CARDOX PRE-INITIATION ALARM actuates.

Which one of the following describes actions, if any, that will occur if only ONE of the sixteen spot type heat detectors in the cable spreading room has reached its setpoint of 140°F?

a. No additional actions will occur. At least two heat detectors must reach their alarm setpoint before any additional actions occur.

b. The Cable Spreading Room A/C unit and exhaust fan will trip. Then, after a 24-second time delay the CO_2 will inject into the room.

c. After a 24-second time delay, the Cable Spreading Room A/C unit and exhaust fan will trip and CO_2 will inject into the room.

d. The Cable Spreading Room A/C unit and exhaust fan trip immediately. CO_2 will inject into the room with NO time delay.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	600000	AK2.01	
	Importance Rating	2.6		

Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors/ detectors, valves

Proposed Question: RO Question # 43

Proposed Answer: B

A: Incorrect – CO2 system actuates, fans trip.

B: Correct – Per SD 513, pages 16-18,
The automatic initiation occurs if ONE of the 16 spot type heat detectors located in the Cable Spreading Room reaches 140°F. At this temperature an electrical signal is sent to the Cardox Control Panel 1C-179. This panel then sends signals to perform the automatic initiation sequence.
The following actions take place upon an automatic initiation:

- Local alarm horn sounds at panel 1C179
- Control room annunciator 1C40 (F-6) CARDOX PRE-INITIATION ALARM actuates
- A 24 second pre-discharge time delay starts
- Cable Spreading Room A/C Unit (1V-AC-32) trips
- Cable Spreading Room Exhaust Fan (1V-EF-33) trips

After the 24 second pre-discharge time delay times out, the following occurs:
CO_2 discharges into the Cable Spreading Room.

C: Incorrect – The A/C unit and fan trips immediately.

D: Incorrect – There is a 24 second time delay before CO2 discharges.

Technical Reference(s): SD 513 Rev 11 pages 16-18; 1C40 (F-6) Rev 64 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X - 19105	
	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	

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

1 Point

44. Which one of the following describes how the Reactor Protection System functions to allow control rod insertion on a valid low RPV level scram signal?

a. RPS energizes the ARI solenoids which repositions the ARI valves to permit venting of the scram air header.

b. RPS trips and back-up scram valves de-energize. This repositions the back-up scram valves to permit venting of the scram air header.

c. RPS trips and scram pilot valves de-energize and reposition to vent air from the scram inlet and outlet valves. This causes the scram inlet and outlet valves to reposition.

d. RPS de-energizes causing a loss of power to the scram inlet and outlet valves. This causes the scram air header to depressurize.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295006	AK2.01	
	Importance Rating	4.3		

Knowledge of the interrelations between SCRAM and the following: RPS

Proposed Question: RO Question # 44

Proposed Answer: C

A: Incorrect – ARI is independent of RPS in regard to venting the scram air header.

B: Incorrect – When RPS trips the back-up scram valves are energized

C: Correct – Per SD 358, page 28 - There are two scram pilot valves for each control rod, arranged as shown in the figure above. Each scram pilot valve is solenoid operated with the solenoids normally energized. The scram pilot valves control the air supply to the respective scram valves for each control rod. With either scram pilot valve energized, air pressure holds the scram valves closed. The scram valves control the supply and discharge paths for control rod scram water. RPS trip system A controls one of the scram pilot valves for each control rod. RPS trip system B controls the other scram pilot valve for each control rod.

D: Incorrect – Scram inlet and outlet valves are air operated. Their position is affected by the scram pilot valve.

Technical Reference(s): SD 358 Rev 7 page 28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

1 Point

45. The plant is at full power.

Then, annunciator 1C07A B-4 “EHC Fluid reservoir 1T-33 LO Level” is received.

The in-plant operator reports an unisolable EHC leak.

The CRS directs Reactor SCRAM and IPOI 5 actions to be carried out.

The CRS enters EOP 1 on low RPV water level after the SCRAM and directs securing both EHC pumps.

As reactor pressure rises, which one of the following describes Bypass Valve operation for this event and subsequent Reactor pressure control?

a. Bypass Valve operation will NOT be available for long term RPV pressure control under these conditions. ADS/SRVs, HPCI, or RCIC can be used for RPV pressure control.

b. The Bypass Valves will NOT control with Pressure Set. However, the Bypass Valve Opening Jack will still function.

c. The installed Bypass Valve accumulators provide 30 minutes of Bypass Valve operation. At this point decay heat will be within the capacity of the MSL Drain Valves.

d. The Bypass Valves will NOT be available for RPV pressure control. Use Chest Warming to control RPV pressure until decay heat is within the capacity of the MSL Drains.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025	EK2.08
	Importance Rating	3.7	

Knowledge of the interrelationships between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system.

Proposed Question: RO Question # 45

Proposed Answer: A

A: Correct - With loss of EHC header pressure the Bypass Valves have about 40 seconds accumulator reserve before the valves go closed. The Bypass Valves will be unavailable for RPV pressure control for EOP 1. Reactor pressure will increase to the high pressure setpoint and LLS will be the pressure control system unless operator action is taken to control pressure.

B: Incorrect - This is a plausible choice because the valves will not respond to pressure set and we do use The Bypass Valve Opening Jack in the EOPs. However, EHC pressure is required.

C: Incorrect - This is a plausible choice because there are accumulators installed in the plant for 30-min. operations. However, this is not the case with loss of EHC header pressure the Bypass valves. They have about 40 seconds accumulator reserve before the valves go closed.

D: Incorrect - Chest warming would reduce pressure. However, with the loss of EHC pressure the valves used for chest warming will not open.

Technical Reference(s): SD 693.2 Rev 7 page 34, EOP 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:			
	Bank #	X	
	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

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

46. Which one of the following is the reason EOP 2, Primary Containment Control, step DWT/4 directs spraying the drywell before drywell temperature reaches 280°F and entering EOP 1?

    a. Reduce RPV Level instrument inaccuracies.

    b. Prevent containment structural failure due to overheating.

    c. Prevent exceeding the environmental qualification of the MSIV solenoids.

    d. Limit the condensation effect of drywell sprays on drywell pressure.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295028	EK3.03	
	Importance Rating	3.6		

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Drywell spray operation: Mark-I&II

Proposed Question: RO Question # 46

Proposed Answer: B

A: Incorrect – RPV Saturation Temperature Graph 1 addresses this concern

B: Correct – Per EOP 2 bases document, page 34 - When drywell temperature is approaching structural design limits in spite of previous temperature control actions, energy release to the drywell is reduced by entering EOP 1 and scramming the reactor.

C: Incorrect – Per The EOP bases, the concern is not the MSIV solenoids.

D: Incorrect – The Drywell Spray Initiation Limit curve addresses this concern

Technical Reference(s): EOP 2 bases Rev 13 page 34 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	WTS	
	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	(9)
	55.43	

(9) Shielding, isolation, and containment design features, including access limitations.

1 Point

47. Which one of the following describes the reason that the initiation of Drywell spray is permitted only within the limits of the Drywell Spray Initiation Limit Curve?

    a. It could result in an evaporative cooling pressure drop large enough to deinert the primary containment atmosphere through the reactor building-to-torus vacuum breakers.

    b. It could result in an evaporative cooling pressure drop large enough to deinert the primary containment atmosphere through the torus-to-drywell vacuum breakers.

    c. To prevent excessive cycling of the reactor building-to-torus vacuum breakers and challenge of the primary containment pressure suppression capability.

    d. To ensure that Suppression Chamber Pressure can be restored below the Torus Spray Initiation Pressure.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295024	EK3.08	
	Importance Rating	3.7		

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Containment spray: Plant-Specific

Proposed Question: RO Question # 47

Proposed Answer: A

A: Correct – EOP 2 bases page 53 discussion of step PC/P-6 - Drywell sprays may only be initiated if drywell temperature and pressure are within the unshaded region of the Drywell Spray Initiation Limit (discussion of the basis for the limit is in the EOP Curves and Limits Bases Document). Initiation of sprays from within the shaded region of the curve could result in an evaporative cooling pressure drop large enough to deinert the primary containment atmosphere through the reactor building-to-torus vacuum breakers or challenge the primary containment negative pressure capability.

B: Incorrect – The concern is with deinerting the primary containment atmosphere through the *reactor building-to-torus* vacuum breakers.

C: Incorrect – Cycling of the breakers is not a concern in the shaded area of the curve.

D: Incorrect – Restoring drywell pressure below the Torus Spray Initiation Pressure may occur but is not the reason for question asked.

Technical Reference(s): EOP 2 bases rev 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	(9)
	55.43	

(9) Shielding, isolation, and containment design features, including access limitations.

1 Point

48. A plant transient has occurred. EOP 1 has been entered and the level control leg is being implemented.

The only low-pressure injection systems available were "A" CS and the "A" RHR pump.

When RPV level lowered to +15", the crew decided that an ED was necessary.

The following plant conditions currently exist:

- "A" Core Spray is injecting at 2800 gpm
- "A" RHR pump is injecting at 4800 gpm
- RPV pressure is 50 psig and lowering
- RPV level is -35" and stable

Which one of following states whether adequate core cooling is assured and the reason for that determination?

a. Adequate core cooling is NOT assured because RPV level is below -25 inches and Core Spray flow is less than 3000 gpm.

b. Adequate core cooling is NOT assured because BOTH the "A" and "B" Core Spray pumps must be injecting at greater than 3000 gpm.

c. Adequate core cooling is assured because sufficient low pressure ECCS injection is occurring.

d. Adequate core cooling is assured because the current RPV level is the lowest level at which steam flow through the SRVs is sufficient to remove all decay heat from the core.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295031	EA1.03	
	Importance Rating	4.4		

Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL : Low pressure core spray

Proposed Question: RO Question # 48

Proposed Answer: A

A: Correct – PER EOP 1 bases page 46 - If RPV water level *cannot* be restored and maintained above -25 inches, adequate core cooling cannot be ensured and SAG entry is required.

B: Incorrect – ONLY one Core Spray Pump at >3000 gpm is required at this RPV level if level is >-39”

C: Incorrect – With Core Spray Pump at 2800 gpm and RPV level is below -25”, adequate core cooling is NOT assured and a SAG entry is required.

D: Incorrect – Reactor pressure is at 50 psig. Steam flow thru the SRVs will not assure adequate core cooling

Technical Reference(s): EOP 1 bases Rev 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Do NOT provide the EOP 1 or ATWS level control legs with setpoints

Learning Objective: (As available)

Question Source:	Bank #	20766	
	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	5, 10
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

49. Plant conditions are as follows:

- EOP-1, RPV Control is being executed.
- High Pressure Coolant Injection (HPCI) is operating for RPV Level control as directed by EOP 1.

Torus Water Level is reported to be 10.1 feet and LOWERING.

Which ONE of the following identifies the Torus Water Level at which HPCI must be secured and the reason it must be secured?

HPCI must be secured when Torus Water Level reaches____.

a. 7.1 feet, to prevent violating Vortex Limits.

b. 7.1 feet, to prevent direct pressurization of the Torus by the HPCI exhaust.

c. 5.8 feet, to prevent violating Vortex Limits.

d. 5.8 feet, to prevent direct pressurization of the Torus by the HPCI exhaust.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030	EA1.05
	Importance Rating	3.5	

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: HPCI

Proposed Question: RO Question # 49

Proposed Answer: D

A: Incorrect – This is a vortex level limit UNLESS directed to use HPCI in EOPs
B: Incorrect – This is above the level that will result in direct pressurization of the torus by the HPCI exhaust.
C: Incorrect – EOP 1 overrides vortex concerns
D: Correct – Per EOP 2 bases step T/L-8 - A torus level of 5.8 feet corresponds to the HPCI turbine exhaust elevation. Direction here attempts to maintain the availability of HPCI should it be needed as an injection source or alternate method of depressurizing the RPV.
Operation of the HPCI system with its exhaust device not submerged will directly pressurize the torus.

Technical Reference(s): EOP 2 bases Rev 13 page 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Do NOT provide EOP 2 Torus Level legs with any setpoints filled in.

Learning Objective: (As available)

Question Source:	Bank #	DAEC	
	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	8, 10
	55.43	

(8) Components, capacity, and functions of emergency systems.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

50. The plant was at full power, with “A” loop of Torus Cooling in service, when a Design Basis Earthquake occurred.

- All control rods were fully inserted.
- “A” ESW pump remains in service.
- Operators are inspecting for flooding damage in the Turbine and Reactor Buildings.
- The plant is currently being cooled down in preparation for going to cold shutdown.

In accordance with AOP-901, EARTHQUAKE, which one of the following actions is required in regard to component cooling water systems?

a. Shutdown General Service Water until a system walkdown to assess damage is complete.

b. RHRSW must be isolated to and from Well Water because the Well Water system is NOT seismically qualified.

c. ESW must be isolated to and from Well Water because the Well Water system is NOT seismically qualified.

d. RHRSW/ESW return must be realigned to the dilution structure because the Circ Water System is NOT seismically qualified.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295018	AA1.03	
	Importance Rating	3.3		

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Affected systems so as to isolate damaged portions

Proposed Question: RO Question # 50

Proposed Answer: C

A: Incorrect – This is not required unless flooding observed.
B: Incorrect – RHRSW does not tie in to well water
C: Correct – Per AOP 901 - Continuous Recheck Statement requires isolation of Well Water and calls the Chill water portion of the Well Water System "non-seismic". ESW operability is still required in Mode 3

D: Incorrect - Circ Water is also non-seismic. This action is directed in case of flooding in the Turbine Bldg. The stem states there is no flooding in the turbine building

Technical Reference(s): AOP 901 Rev 17 page 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	DAEC	
	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	

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

1 Point

51. The plant is operating at full power. All systems are operable.

The River Water Supply (RWS) system is in operation with BOTH of the B RWS Subsystem Pumps, 1P-117B, B RWS Pump, and 1P-117D, D RWS Pump, running in AUTO.

RWS HSS-2911B "LOAD SHED AUTO START RWS PUMP SELECT" is selected to 1P-117B.

Then, a loss of offsite power occurs and the B SBDG restores power to bus 1A4 as designed.

Which of the following describes the response of the B and D RWS Pumps to this transient?

a. Immediately after bus 1A4 restoration, the B RWS Pump will automatically start. D RWS Pump will not automatically start.

b. 2 minutes after bus 1A4 restoration, both the B and D RWS Pumps will automatically start.

c. Immediately after bus 1A4 restoration, the B RWS Pump will automatically start. 2 minutes after bus 1A4 restoration, the D RWS Pump will automatically start.

d. 2 minutes after bus 1A4 restoration, the B RWS Pump will automatically start. D RWS Pump will not automatically start.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295003	AA2.04	
	Importance Rating	3.5		

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : System lineups

Proposed Question: RO Question # 51

Proposed Answer: D

A: Incorrect – B Pump will auto start after a 2 minute time delay.

B: Incorrect - Even though it is in AUTO, the D pump will not auto start.

C: Incorrect – B Pump will auto start after a 2 minute time delay. Even though it is in AUTO, the D pump will not auto start.

D: Correct - Per SD 410, page 8, There are two Load Shed Auto Start RWS Pump Select switches. HSS-2911A is a selector switch for Pump 'A' or Pump 'C' and HSS-2911B is a selector switch for Pump 'B' or Pump 'D'. The non-operating RWS pump in each subsystem is normally selected. The selected pump will automatically restart following a loss of offsite power. If the selected pump was not running prior to the power loss, it will start immediately when power is regained from respective emergency diesel. If the selected pump was previously running, it will restart after the 2 minute time delay is satisfied.

Technical Reference(s): SD 410 Rev 8 page 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	20625	
	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	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

52. The plant was operating at full power when an event occurred. The operators scrammed the plant and placed the Mode Switch in SHUTDOWN. The following describes the control rod status.

- Control rod 18-23 is at position “44”
- 20 other control rods are at position “02”
- All other control rods are at position “00”

Which one of the following describes the status of the reactor?

The reactor is __(1)__under ALL conditions because __(2)__

a. (1) NOT Shutdown
   (2) control rod 18-23 is NOT at its Maximum Subcritical Banked Withdrawal Position.

b. (1) NOT Shutdown
   (2) 21 control rods are NOT at their Maximum Subcritical Banked Withdrawal Position.

c. (1) Shutdown
   (2) ONLY one control rod is NOT at its Maximum Subcritical Banked Withdrawal Position.

d. (1) Shutdown
   (2) adequate shutdown margin exists in the core to account for the control rods NOT at position “00.”

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295037	EA2.05	
	Importance Rating	4.2		

EA2.05 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Control rod position

Proposed Question: RO Question # 52

Proposed Answer: B

A: Incorrect – The reactor is not shutdown because more than one control rod is not at its Maximum Subcritical Banked Withdrawal Position

B: Correct – Per ATWS EOP bases page 4 - All control rods inserted to at least position 00 is the Maximum Subcritical Banked Withdrawal Position. It is defined as the lowest control rod position to which all control rods may be withdrawn in a bank and the reactor will nonetheless remain shutdown under all conditions, irrespective of reactor coolant temperature and any boron, which may have been injected into the RPV. "Shutdown under ALL conditions without boron" can be determined by relying on the Technical Specification demonstration of adequate shutdown margin:

- One control rod is out beyond position 00
- All other control rods are at position 00

C: Incorrect – The reactor is not shutdown. 21 control rods are not at their Maximum Subcritical Banked Withdrawal Position

D: Incorrect – The reactor is not shutdown. Adequate shutdown margin exists for only one rod not at its Maximum Subcritical Banked Withdrawal Position

Technical Reference(s): ATWS EOP Bases Rev 12 Page 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5
	55.43	

53. The plant is operating at 90% power.

Due to thunderstorms in the area causing grid instability conditions, the load dispatcher requests increasing Megavars from 100 to 250.

Given the attached Generator Reactive Capability Curve and the following information:

- Hydrogen Pressure = 30 psig
- Generator Megavars Lagging = 100
- Megawatts = 600

Determine what, if any, curve limitation will be exceeded if the Megavars are increased as requested.

a. NO curve limitation will be exceeded.

b. The curve limitation for Field heating will be exceeded.

c. The curve limitation for Armature heating will be exceeded.

d. The curve limitation for Armature Core End heating will be exceeded.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000	AA2.01
	Importance Rating	3.5	

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operating point on the generator capability curve.

Proposed Question: RO Question # 53

Proposed Answer: B

A: Incorrect – The curve limitation for Field heating will be exceeded.

B: Correct – Per Generator Reactive Capability Curve in OI 698 App.1, The curve limitation for Field heating will be exceeded.

C: Incorrect – The curve limitation for Field heating will be exceeded.

D: Incorrect - The curve limitation for Field heating will be exceeded.

Technical Reference(s): OI 698 App.1, Rev 70 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OI 698 App.1

Learning Objective: (As available)

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
55.43

(4) Secondary coolant and auxiliary systems that affect the facility.

54. The plant was operating at full power when a loss of feedwater heating occurred.

The plant has been scrammed. Indications of fuel damage exist.

The following annunciators are in alarm:

- 1C05B (C-2) MAIN STEAM LINE HI HI RAD / INOP TRIP
- 1C05B (D-2) MAIN STEAM LINE HI RAD
- 1C03A (A-4) OFFGAS VENT PIPE RM-4116A/B HI-HI RAD

The CRS has entered AOP 672.2 “Offgas Radiation/Reactor Coolant Activity High”.

Which one of following describes actions that will automatically occur and other required manual actions?

a. All MSIVs, the Main Steam Line Drain Valves and the Recirc Sample CVs will automatically close.
SBGT will automatically start.

b. The Main Steam Line Drain Valves and the Recirc Sample CVs will automatically close.
The MSIVs will NOT automatically close and must be manually closed.
SBGT will NOT automatically start.

c. The Main Steam Line Drain Valves and the Recirc Sample CVs will automatically close.
SBGT will automatically start.
The MSIVs will NOT automatically close and must be manually closed.

d. The Main Steam Line Drain Valves will automatically close.
SBGT will NOT automatically start.
The MSIVs will NOT automatically close and must be manually closed.
The Recirc Sample CVs will NOT automatically close.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 2.1.23	
	Importance Rating	4.3	

(High Offsite release rate) Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: RO Question # 54

Proposed Answer: C

A: Incorrect – The MSIVs must be closed manually, they do not auto close on steam line high rad.

B: Incorrect - SBGT auto starts on a Group 3 signal

C: Correct – Per AOP 672.2 automatic action section and followup action step 6. The Main Steam Line Drain Valves and the Recirc Sample CVs should have closed. SBGT should have started. The MSIVs must be manually closed.

D: Incorrect - The Recirc Sample CVs auto close. SBGT auto starts on a Group 3 signal

Technical Reference(s): AOP 672.2 Rev 33 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7,10
	55.43	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

1 Point

55. The plant was operating at full power when a fire in the control room required the control room to be abandoned.

You have been dispatched to 1C208 to operate RCIC to control reactor water level.

Which one of the below describes RCIC operations that can be performed at panel 1C208 and the location where RPV level may be monitored?

a. Control of RCIC TURB SPEED CONTROLLER HIC-2440 ONLY with no local speed or flow indication.
RPV may be monitored on panel 1C388, Remote Shutdown Panel.

b. Control of RCIC INJECTION VALVE MO-2512 and RCIC TURB SPEED CONTROLLER HIC-2440 with no local speed or flow indication.
RPV may be monitored on panel 1C388, Remote Shutdown Panel.

c. Control of TURBINE STEAM SUPPLY MO-2404 and RCIC TURB SPEED CONTROLLER HIC-2440 with local flow indication ONLY.
RPV level may be monitored on RCIC Panel 1C208.

d. Control of RCIC TURB SPEED CONTROLLER HIC-2440 ONLY with no local speed or flow indication.
RPV level may be monitored on RCIC Panel 1C208.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295016	AA1.06	
	Importance Rating	4.0		

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Reactor Water Level

Proposed Question: RO Question # 55

Proposed Answer: A

A: Correct – Per AOP 915 page TAB2 note prior to step 2, RCIC can be controlled from 1C208 with HIC-2440 but no flow or speed indication is available.

B: Incorrect – ONLY RCIC turbine speed with HIC-2440 can be controlled at the remote panel.

C: Incorrect - RCIC can be controlled from 1C208 with HIC-2440 ONLY. There is no panel 1C208 indication for RCIC speed, flow or RPV level.

D: Incorrect - RCIC can be controlled from 1C208 with HIC-2440 ONLY. There is no panel 1C208 indication for RCIC speed, flow or RPV level.

Technical Reference(s): AOP 915 Rev 39 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	19072	
	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	

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

1 Point

56. During a prolonged loss of the instrument and service air system, the operator is directed to maintain reactor water level in the normal operating band using a specific method stated in AOP 518, Failure of Instrument and Service Air.

    What is the method and the reason for its use?

    a. Throttle the A and B FEEDLINE BLOCK valves MO-1636 and MO-1592 if the Feed Reg Valves lockup.

    b. Throttle the A and B FEEDLINE BLOCK valves MO-1636 and MO-1592 due to the possibility of the Feed Reg Valves drifting CLOSED.

    c. Throttle the Feed Reg Valves because the FEEDLINE BLOCK valves MO-1636 and MO-1592 will NOT properly operate due to excessive differential pressure across the valves with a Feedwater pump running.

    d. Throttle the Feed Reg Valves because the FEEDLINE BLOCK valves MO-1636 and MO-1592 will have locked up.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295019	2.1.20	
	Importance Rating	4.6		

(Partial or total loss of Inst Air) Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 56

Proposed Answer: A

A: Correct – Per AOP 518 Note & Caution prior to step 10. - During a prolonged loss of air casualty the Feed Reg valves may drift open. If the Feed Reg Valves lock up, Maintain Reactor water level in the normal operating band by throttling A and B FEEDLINE BLOCK valves MO-1592 and MO-1636

B: Incorrect – the FRVs would drift OPEN

C: Incorrect – This is not specified in the AOP. The block valves may not REOPEN due to excessive DP with a Feedwater pump running.

D: Incorrect – The Feedline Block valves are not air operated and will not lockup

Technical Reference(s): AOP 518 Rev 31 page 4 & 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #	19511 19512	(Note changes or attach parent)
	New		

ORIGINAL QUESTION

Due to lowering Instrument Air Pressure, AOP-518, FAILURE OF INSTRUMENT AND SERVICE AIR is being executed.

1C05A, F-1, “A” or “B” FEED REG VALVE POSITION LOCKED is activated.

If Instrument Air pressure cannot be restored, how are Feedwater Regulating Valves, CV-1579 and CV-1621, affected; and what procedural action(s) is (are) required?

Feedwater Regulating Valves will fail:

_____ A. SHUT; it is required to reduce Reactor Power to control RPV Water Level.

_____ B. SHUT; it is required to use the FEEDWATER STARTUP CONTROL VALVE, CV-1622 to control RPV Water Level.

_____ C. OPEN; it is required to THROTTLE A AND B FEEDLINE BLOCK Valves, MO-1592 and MO-1636 to control RPV Water Level.

_____ D. OPEN; it is required to completely SHUT A FEEDLINE BLOCK Valve, MO-1592, OR B FEEDLINE BLOCK Valve, MO-1636, to control RPV Water Level

Question History: Last NRC Exam: 2007

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

1 Point

57. The reactor was shutdown six days ago for a Refueling Outage. Reactor Coolant System Temperature is 150°F. Core Alterations have NOT been performed.

The cavity is flooded up and the fuel pool gates are removed.

With NO Decay Heat Removal, how long will it take for Reactor Coolant System Temperature to reach 200°F using the provided reference?

a. <2 hours

b. 5 hours

c. 10 hours

d. 12 hours

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295021	2.1.25	
	Importance Rating	3.9		

(Loss of Shutdown Cooling) Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 57

Proposed Answer: C

A: Incorrect – would be correct if Appendix 2 was used in error
B: Incorrect – would be correct if Appendix 3 was used in error
C: Correct - Using AOP-149 Appendix 1 Curve, 5°F / hr heatup rate should be obtained.
200°F - 150°F = 50°F 50°F / 5 °F / hr = 10 hr using Appendix 1 of AOP 149, a 5 degree per hour heatup rate is obtained. To rise from 150 to 200 degrees is a 50 degree change. At 5 degrees per hour that would take 10 hours
D: Incorrect – would be correct if initial temperature was 100 degrees and Appendix 1 was used in error

Technical Reference(s): AOP 149 Rev 31 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: AOP 149 Appendices 1,2 and 3

Learning Objective: (As available)

Question Source:	Bank #	X	
	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

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

1 Point

58. The reactor scrammed and the MSIVs have closed due to a small break in the piping from the Main Steam Line Equalizing Header.

SRVs are now being cycled to control reactor pressure. Suppression Pool level has risen to 13.8 feet.

If Suppression Pool level cannot be restored and maintained below 13.8 feet, Emergency Depressurization is required because _____

a. Suppression Pool level is approaching the Safety Relief Valve Tailpipe Vacuum Breakers.

b. the containment spray ring header is completely submerged and containment integrity may be compromised.

c. continued SRV operation may cause tailpipe damage and directly pressurize containment.

d. a large break LOCA will result in drywell pressure exceeding design due to Suppression Pool level approaching the Torus-to-Drywell vacuum breaker level.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295025	EA2.04	
	Importance Rating	3.9		

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool level

Proposed Question: RO Question # 58

Proposed Answer: C

A: Incorrect – These are located in the drywell
B: Incorrect – The level of the bottom of the ring header is 13.83 ft
C: Correct – Per EOP 2 Bases page 16 - the SRV Tail Pipe Level Limit is the highest torus water level at which opening of an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, T-quencher, or T-quencher supports. The SRV Tail Pipe Level Limit is a function of torus water level and RPV pressure. SRV operation with torus water level above the SRV Tail Pipe Level Limit could damage the SRV discharge lines. This, in turn, could lead to containment failure from direct pressurization and damage to equipment inside the containment (ECCS piping, RPV water level instrument runs, torus-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.
D: Incorrect – This level is at 13.5 ft and Emergency Depress is not required at that point

Technical Reference(s): EOP 2 Bases Rev 13 page 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	WTS 2468	
	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	3
	55.43	

(3) Mechanical components and design features of the reactor primary system.

1 Point

59. The plant is at full power. A loss of the running CRD pump has occurred.

    Which one of the following describes the motive force to insert the control rods if accumulator pressures fall too low to accomplish the task?

    a. A ball check valve, located in the CRDM cylinder flange inlet port, repositions and allows reactor water pressure to act on the CRDM under piston area.

    b. Cooling Water header pressure unseats a ball check valve, located in the CRDM cylinder flange inlet port, creating a flow path for reactor water through the drive header to the CRDM under piston area.

    c. A ball check valve, located in the CRDM cylinder flange withdraw port, repositions and allows reactor water pressure to act on the CRDM under piston area.

    d. Ball check valves, located in the CRDM cylinder flange inlet and withdraw ports, reposition and allow reactor water pressure to act on the CRDM under piston area.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295022	AK1.01	
	Importance Rating	3.3		

Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: Reactor pressure vs. rod insertion capability

Proposed Question: RO Question # 59

Proposed Answer: A

A: Correct – Per SD 255, page 46 - As water is forced from the accumulator, accumulator pressure falls below reactor pressure and causes the ball check valve (located in the insert port) to shift its position. This admits reactor pressure into the under piston area, completing the scram stroke.

B: Incorrect – Reactor pressure unseats the ball check valve

C: Incorrect – the ball check is located in the cylinder flange inlet port

D: Incorrect – there is only one ball check in the CRDM

Technical Reference(s): SD 255 Rev 8 page 46 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 10.07.01.06-07 (As available)

Question Source:	Bank #	19991	
	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	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

60. The plant was operating at full power when an event occurred. The following conditions exist:

- Reactor Level is 58” and lowering slowly
- Reactor pressure is 750 psig and lowering slowly
- Drywell Pressure is 3.5 psig and rising slowly
- Drywell temperature is 152°F and rising slowly
- HPCI and RCIC are in service
- All Low Pressure ECCS pumps are running
- Well Water is in service

Which one of the following actions, if any, is required to maximize drywell cooling IAW EOPs?

a. The Drywell Cooling Loop A and B MODE SELECT hand switches HS-5718A and B must be taken to INOPERATIVE and returned to START. This will reset the 2 psig seal-in and allow the fans to shift to fast speed.

b. Install Defeat 4. Well Water must be shutdown then restarted.

c. Install Defeat 4. Well water does NOT need to be secured.

d. No switch manipulations are required. Drywell Cooling Fans will have shifted to fast speed on the high drywell pressure signal.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295012	AK2.02	
	Importance Rating	3.6		

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling

Proposed Question: RO Question # 60

Proposed Answer: B

A: Incorrect – Defeat 4 must be installed due to the PCIS Group 7 isolation on lo-lo-lo RPV level

B: Correct - due to the PCIS Group 7 isolation on lo-lo-lo RPV level Defeat 4 must be installed and well water must be secured prior to installation to prevent water hammer. (See Defeat 4 installation document)

C: Incorrect – Well water must be secured

D: Incorrect – Defeat 4 must be installed

Technical Reference(s): Defeat 4 Rev 8 page 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (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

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

1 Point

61. Which one of the following describes the reason for the Recirc pump runback.

The __(1)__ Recirc pump runback ___(2)___

a. (1) 20%
   (2) allows additional time for operator action prior to reaching the low level scram setpoint.

b. (1) 20%
   (2) ensures the remaining Reactor Feed Pump is NOT operated at run-out conditions.

c. (1) 45%
   (2) allows additional time for operator action prior to reaching the low level scram setpoint.

d. (1) 45%
   (2) ensures the remaining Reactor Feed Pump is NOT operated at run-out conditions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009	AK3.01
	Importance Rating	3.2	

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

Proposed Question: RO Question # 61

Proposed Answer: C

A: Incorrect – The 20% limiter is to prevent Recirc pump and jet pump cavitation.
B: Incorrect – The 20% limiter is to prevent Recirc pump and jet pump cavitation.
C: Correct per SD 264 - Limiter #2 will act to reduce reactor power under conditions where the reactor water level is decreasing due to a possible malfunction in the Feedwater System such as the tripping of a Feedwater pump. The reduction in reactor power will slow the rate of reactor water level decrease and allow additional time for operator action prior to reaching the low level scram setpoint. Limiter #2 allows a maximum speed control signal of 45%.

D: Incorrect – Per the SD, ensuring pump runout is not the concern..

Technical Reference(s): SD 264 Rev 11 page 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 12.00.00.02 (As available)

Question Source: Bank #
Modified Bank # 2005 NRC (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 5
55.43

. ORIGINAL QUESTION from 2005 NRC exam #23 – correct answer “A”

Which one of the following describes the reason for the Recirc pump runback.
The __(1)__ Recirc pump runback ___(2)___.

a. (1) 20%
(2) protects the recirc pumps against cavitation due to low feedwater flow
b. (1) 20%
(2) prevents damage to the Recirc pump thrust bearing due to increased axial thrust when the Recirc Pump Discharge Bypass Valve is CLOSED
c. (1) 45%
(2) ensures the 170” scram on RVP level will not occur from rated full power.
d. (1) 45%
(2) ensures the remaining Reactor Feed Pump is NOT operating at run-out conditions.

1 Point

62. A plant event has occurred.

With HPCI operating in the Pressure Control Mode with the Flow Indicating Controller (FIC) in automatic, Reactor Pressure is at 1050 psig and slowly rising.

Which ONE of the following will cause Reactor Pressure to lower?

(Assume the ONLY effect on RPV pressure is HPCI operation)

a. Shut MO-2202, TURBINE STEAM SUPPLY VALVE.

b. Adjust CV-2315, TEST BYPASS VALVE from 47% to 55% open.

c. Adjust CV-2315, TEST BYPASS VALVE from 47% to 40% open.

d. Adjust FIC-2309, HPCI FLOW CONTROL from 3000 gpm to 2600 gpm.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295007	AA1.02	
	Importance Rating	3.5		

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE : HPCI

Proposed Question: RO Question # 62

Proposed Answer: C

A: Incorrect – this would reduce steam flow to the HPCI turbine and not lower RPV pressure

B: Incorrect – This would cause HPCI to pump at a lower discharge head requiring less energy. Therefore RPV pressure would rise

C: Correct - Throttling shut CV-2315, TEST BYPASS VALVE from 47% to 40% Open, will cause HPCI to pump the same recirculation flow at a higher discharge head, which consumes MORE energy from the HPCI Turbine, requiring more Main Steam flow to the HPCI Turbine, which lowers Reactor Pressure.

D: Incorrect – lowering speed will reduce the energy required to operate the HPCI turbine. Wth less energy required for HPCI, RPV pressure would rise.

Technical Reference(s): OI 152 Rev 93, discussion – QRC 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

63. The plant is shutdown for refueling with the following conditions:

- Core Alterations are in progress.
- The Refueling Supervisor reports that a fuel bundle has been loaded into the wrong reactor core location.
- The Control Room operator observes that Source Range count indication for the SRM in that quadrant, has increased and stabilized at a higher value.

As a result of this event, Shutdown Margin (SDM) has __(1)__ and the reactor __(2)__.

a. (1) increased
   (2) remains subcritical

b. (1) increased
   (2) is critical

c. (1) decreased
   (2) remains subcritical

d. (1) decreased
   (2) is critical

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295014	AA2.01	
	Importance Rating	4.1		

Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION : Reactor power

Proposed Question: RO Question # 63

Proposed Answer: C

Explanation: The KA matches because any inadvertent reactivity addition and its relation to reactor power is contingent on reactor theory knowledge

A: Incorrect - SDM decreases anytime fuel is added to the core.

B: Incorrect - SDM decreases anytime fuel is added to the core. Counts would be increasing if the reactor was supercritical.

C: Correct - SDM decreases anytime fuel is added to the core. Since count stabilized in the SR the reactor is still sub-critical

D: Incorrect - Counts would be increasing if the reactor was supercritical.

Technical Reference(s): TS Definition
GFES Rx Theory Chap 2 pg 20 (Attach if not previously provided)
GFES Rx Theory Chap 3 pg 8

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	1
	55.43	

(1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

1 Point

64. The plant has been operating at 90% thermal power for several days. Over the last several hours the following conditions have changed:

- Condenser Backpressure has risen from 2.5” Hg to 3.5” Hg
- Offgas system flow has lowered from 20 scfm to 18 scfm
- MWE lower by 10 MW
- There have been NO alarms received

Which one of the following could be the cause of these indications?

a. Air leak into the condenser.

b. Offgas premature recombination.

c. One or more blown Offgas loop seals.

d. Cooling tower outlet temperature increased.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002	2.4.47
	Importance Rating	4.2	

(Loss of Main Condenser Vacuum) Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: RO Question # 64

Proposed Answer: D

A: Incorrect – An air leak would cause Backpressure to increase but would also cause Offgas flow to increase.

B: Incorrect – Premature recombination would cause Offgas flow to decrease but not Backpressure increase. There would also be several alarms associated with this problem.

C: Incorrect - Blown loop seals would cause Offgas flow to decrease but not Backpressure increase. There would also be radiation alarms associated with this problem.

D: Correct - The rise in cooling tower outlet temperature causes less condensate subcooling, the result is lower plant efficiency causing the lowering of MWE and higher circ water temp will cause less condensing of steam resulting in a higher condenser backpressure. Less vacuum in the condenser will cause less air inleakage lowering the Offgas system flow.

Technical Reference(s): ARP 1C07B (D-9) rev 69
AOP 672.3 Rev 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	X	
	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	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

65. What is the reason for automatic closure of the RWCU Primary Containment Isolation Valves if a RWCU area high temperature were to occur.

    What is the reason for this requirement?

    a. To ensure that the release of radioactive material to the environment will be consistent with the assumptions used in the final safety analyses.

    b. To minimize moisture buildup and overheating in the Standby Gas Treatment System charcoal beds.

    c. To prevent exceeding the Environmental Qualification temperature limits on the electrical buses in the Turbine Building required for safe shutdown.

    d. To ensure operator access to secondary containment for event mitigation actions.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295032	EK3.03	
	Importance Rating	3.8		

Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Isolating affected systems

Proposed Question: RO Question # 65

Proposed Answer: A

A: Correct – Per TS bases 3.3.6.1. - The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs). Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

B: Incorrect – This would be a concern but is not the reason associated with the TS required PCIV isolation.

C: Incorrect – the reason is to limit the environmental release and not for EQ concerns.

D: Incorrect – the reason is to limit the environmental release, not personnel access

Technical Reference(s): TS bases 3.3.6.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: Objective – 50007.05.06 – Based on information presented or reviewed in the classroom, material in the system description, plant drawings and operations procedures and when given a written exam, state the applicability of the DAEC TS to the PCIS instrumentation and Primary Containment Isolation valves, including the bases for any applicable LCOs without error. (As available)

Question Source:	Bank #	WTS 1326	
	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	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

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

In accordance with ACP 110.1, Conduct of Operations, when an operating crew correctly employs a conservative decision-making policy, all crewmembers understand that when faced with unexpected or uncertain plant conditions, they must __(1)__, and that all decisions must be based on maintaining __(2)__.

a. (1) place the plant in a safe condition
   (2) nuclear and industrial safety

b. (1) place the plant in a safe condition
   (2) nuclear safety ONLY

c. (1) reduce power or scram the reactor
   (2) nuclear and industrial safety

d. (1) reduce power or scram the reactor
   (2) nuclear safety ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.39	
	Importance Rating	3.6	

Conduct of Operations: Knowledge of conservative decision making practices.

Proposed Question: RO Question # 66

Proposed Answer: A

Explanation (Optional): The KA is matched because the operator must demonstrate knowledge of conservative decision making practices, specifically, its principle (decisions are made with regard to both Nuclear and Industrial Safety) and expectations (the goal is to place the plant in a safe operating condition).

A. Correct - According to SOER 94-1 (p26; Rev 1), which is referenced by LP 50007-93.16 (p5; Rev 16), the key message in conservative decision-making is the expectation that operators, when faced with unexpected or uncertain conditions, must place the plant in a safe condition, and must not hesitate to reduce power or scram the reactor. DAEC, using Attachment 5 of ACP 110.1, has expanded the message and the process to include both Nuclear and Industrial Safety. According to ACP 110.1 (p5; Rev 17) Step 5.1.1, plant operations shall be conducted in a manner that establishes nuclear and personal safety as the highest priority while employing a conservative decision making process. According to ACP 110.1 (p20-21; Rev 17) Attachment 5, personal safety is stated as industrial safety. The principle statement of conservative decision-making policy identifies that Nuclear and industrial safety is maintained at the forefront of all decisions. Based on stated expectations of Attachment 5, the overall philosophy of conservative decision-making must be summed up by stating that when unexpected or uncertain plant conditions, operators must place the plant in a safe condition, rather than the more narrow approach of reducing power or scramming the reactor, although these strategies may be a part of such as policy. For instance, LP 50007-94.24 (p7; Rev 7) instructs the operator to consider the conservative decision-making policy of ACP-110.1, Attachment 5, in regard to sending individuals to perform outside Inspections during severe weather events such as a tornado or a thunderstorm. This clearly indicates that the DAEC policy is more broad based than SOER 94-1 and includes a focus on both nuclear and industrial safety.

B. Incorrect - 1st part correct, 2nd part wrong. This is plausible because SOER 94-1 was written to emphasize nuclear safety.

C. Incorrect - 1st part wrong, 2nd part correct. This is plausible because SOER 94-1 specifically states that the operators when making conservative decisions with respect to nuclear safety should not hesitate to reduce power or scram the reactor.

D. Incorrect - 1st part wrong, 2nd part wrong. This is plausible because SOER 94-1 specifically states that the operators when making conservative decisions with respect to nuclear safety should not hesitate to reduce power or scram the reactor, and SOER 94-1 was written to emphasize nuclear safety.

Technical Reference(s):	SOER 94-1 (p26; Rev 1) LP 50007-93.16 (p5; Rev 16) ACP 110.1 (p5, 20-21; Rev 17) LP 50007-94.24 (p7; Rev 7)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective:	LP 50007 96.07, Objectives 96.07.04.01 & 96.07.04.03	(As available)

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

Question History: NA Last NRC Exam: NA

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

1 Point

67. The plant is operating at full power. All systems are operable.

Which one of the following describes how the Offgas & Recombiner System functions to create a substantial reduction in the release of radioactive materials to the environment?

a. The system reduces the volume of Offgas flow; AND
Delays the release of Hydrogen and Oxygen to the environment.

b. The system reduces the volume of Offgas flow; AND
Delays the release of Xenon and Krypton to the environment.

c. The system recombines short-lived radioactive gases; AND
Delays the release of Hydrogen and Oxygen to the environment.

d. The system recombines short-lived radioactive gases; AND
Delays the release of Xenon and Krypton to the environment.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.27	
	Importance Rating	3.9	

Conduct of Operations: Knowledge of system purpose and/or function.

Proposed Question: RO Question # 67

Proposed Answer: B

A. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that it is the Hydrogen and Oxygen that undergoes a delayed release.

B. **Correct.** 1st part correct, 2nd part correct. According to SD-672 (p5-7; Rev 13), the Offgas and Recombiner System is functionally two subsystems: the recombiner subsystem and the charcoal adsorber subsystem. The recombiner reduces the total volume of the offgas flow by recombining radiolytically dissociated hydrogen and oxygen to produce water vapor. After recombination, the offgas flow consists of small volume amounts of fission product and activation gases carried in the airflow arising out of inleakage to the condenser. This offgas stream is delayed for decay of short lived radioactive isotopes, and then conditioned to a low moisture content and the proper temperature for maximum delay in the charcoal adsorber system. The long holdup time produced by the charcoal adsorbers permits the xenon and krypton gases to decay to particulate daughter products, which either remain on the charcoal or are removed by high efficiency particulate (HEPA) filters. The composite effect of the <u>reduction in system gas volume</u> and the <u>delay produced by charcoal adsorption</u> results in a substantial reduction in the release of radioactive materials to the environment.

C. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that it is the short lived activation gases (i.e. N-16, O-19, and N-13, which are all gases expected to be within the system) that are recombined, and that it is the Hydrogen and Oxygen that undergoes a delayed release.

D. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator may incorrectly believe that it is the short lived activation gases (i.e. N-16, O-19, and N-13, which are all gases expected to be within the system) that are recombined.

Technical Reference(s): SD-672 (p5-7; Rev 14) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: LP 50007 47.0 Objective 47.00.00.01 (As available)

Question Source: Bank # 22304

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

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1 Point

68. While performing a step in a Surveillance Test Procedure, it has been determined that a normally open, motor operated Primary Containment Isolation Valve will not stroke in the closed direction, as required by the procedure.

Which one of the following identifies when the Technical Specification LCO action time is started?

The LCO time would start_____

a. when the surveillance is logged as complete.

b. as soon as the valve failure was recognized.

c. when the start of the surveillance was logged on.

d. at the time the surveillance was last satisfactorily completed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.12	
	Importance Rating	3.7	

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: RO Question # 68

Proposed Answer: B

A. Incorrect – the LCO is entered upon discovery of the issue

B. Correct – IAW TS 3.0.2 - Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

C. Incorrect – the LCO is entered upon discovery of the issue

D. Incorrect – the LCO is entered upon discovery of the issue

Technical Reference(s): TS LCO 3.0.2. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: 1.07.03.04 (As available)

Question Source: Bank # X – NMP 2005
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

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

1 Point

69. The plant is operating at full power.

At the start of your shift (0700) the following conditions exist.

- Annunciator 1C-06A (A-1) - "A" RWS PIT LO LEVEL is in alarm and is LIT due to scheduled ongoing preventive maintenance. The annunciator became LIT when work began 12 hours ago. The work will be completed in 6 more hours.
- Annunciator 1C-06A (B-5) - "A" COOLING TOWER HI/LO LEVEL alarmed at 0000 and is LIT due a failed instrument. Maintenance will complete the work, including retests, in 8 hours.

Which one of the following describes how these alarms are tracked IAW ACP 1410.12, Operator Burden Program?

(Assume that maintenance will complete the work and retests as scheduled)

a. Neither annunciator is required to be entered in the Operator Burden Database.

b. BOTH annunciators are required to be entered in the Operator Burden Database by 1200.

c. ONLY the "A" COOLING TOWER HI/LO LEVEL annunciator is required to be entered in the Operator Burden Database by 1200.

d. ONLY the "A" RWS PIT LO LEVEL annunciator is required to be entered in the Operator Burden Database immediately.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.43	
	Importance Rating	3.0	

Equipment Control: Knowledge of the process used to track inoperable alarms.

Proposed Question: RO Question # 69

Proposed Answer: C

A. Incorrect – The "A" COOLING TOWER HI/LO LEVEL must be entered because it was due to a failed instrument

B. Incorrect – The "A" RWS PIT LO LEVEL is not required to be entered due to work being done for preventive maintenance

C. Correct – Per the ACP, The "A" COOLING TOWER HI/LO LEVEL must be entered because it was due to a failed instrument

D. Incorrect - The "A" RWS PIT LO LEVEL is not required to be entered due to work being done for preventive maintenance

Technical Reference(s): ACP 1410.12 Rev 16 – Definition 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

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

Question History: None Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

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

1 Point

70. Which one of the following activities requires that a plant area be evacuated and a normally unlocked room be locked to prevent plant personnel from being over-exposed to radiation during the activity?

   a. Conducting TIP traces while at power.

   b. Conducting a planned cold start of the HPCI Pump.

   c. Isolating RWCU while at power.

   d. Shifting the Spent Fuel Cooling Filter Demineralizer from the Filter to the Hold Mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.12	
	Importance Rating	3.2	

Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO Question # 70

Proposed Answer: A

A. **Correct.** According to SD-878.6 (p15; Rev 6), due to exposure to the high neutron flux in the core during detector operations, the TIP becomes highly radioactive through activation reactions. To provide protection for personnel, the probe is retracted for storage in a shielded chamber. After the use of the TIP during reactor power operation, radiation levels in the vicinity of the chamber shield may be in the range of 200-500 mRem/hr. According to OI 878.6 (p8; Rev 39), Step 4.1.2, prior to conducting TIP traces at power, the operator must contact Health Physics to ensure there are no personnel in the TIP Machine Room and close and lock the door or ensure the TIP Drive Machine Room Door is under control of Health Physics.

B. **Incorrect.** According to OI 152 (7; Rev 93), access to the HPCI Room shall be controlled during a planned cold start until steady state operating conditions are achieved. Personnel may be present in the room, if necessary, to 1) locally manipulate valves or controls; or 2) perform troubleshooting which cannot be performed in a practical manner from outside the room. Otherwise, prior authorization from the Control Room Supervisor is required. While the access to the room is controlled during planned cold pump starts, it is NOT controlled to minimize radiation exposure, but rather for industrial safety reasons. According to LP 50007_5.0 (p21; Rev 10), during a quarterly surveillance test of the HPCI system at the Quad Cities Station, the exhaust line rupture disc on the Unit One HPCI turbine burst, releasing steam into the HPCI room, burning, and slightly contaminating, five workers. At the DAEC, access to the HPCI and RCIC rooms is controlled during planned cold starts. If an individual must be present in the room during a planned cold start, that individual must contact the OSS for any special precautions which must be taken prior to gaining access to the room. Planned starts of equipment are routinely announced over the plant paging system. Operators must evaluate each task for its possible consequences with respect to personal safety.

C. **Incorrect.** According to OI 261 (13; Rev 79), when RWCU is isolated, the alternate sample line dose rates raise by a factor of 10. This line is located near the entrance to the RWCU Pump room. The increased dose rates affect the Security Post stationed near that sample line. Because of this, when the RWCU System is isolated Health Physics must be notified that dose rates near RWCU pump room door will be elevated and to coordinate with Security to relocate officers, as allowed. However, the room is NOT isolated to control exposure to individuals, also RWCU HX room normally locked.

D. **Incorrect.** According to OI 435 (p24; Rev 55), when backwashing the SPC Filter Demineralizer in accordance with Section 7.7, The HP Shift Technician may restrict access to the Jungle Room during Fuel Pool bed backwashes due to radiation levels, the room is not locked. While this procedure includes going to the Hold Mode of operation as a prerequisite, it does NOT by itself require that access to specific areas of the plant be restricted for the personal exposure control.

Technical Reference(s):
SD-878.6 (p15; Rev 6)
OI 878.6 (p8; Rev 39), Step 4.1.2
OI 152 (7; Rev 93)
LP 50007_5.0 (p21; Rev 10)
OI 261 (13; Rev 79)
OI 435 (p24; Rev 55)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: 50007_83.0, Objective 83.01.01.02 (As available)

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

Question History: New Last NRC Exam: NA

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

10 CFR Part 55 Content:		
	55.41	12
	55.43	

(12) Radiological safety principles and procedures.

1 Point

71. DAEC is in a refueling outage with Fuel Shuffle #1 completed.

- The refueling cavity is flooded
- The fuel pool gates are removed
- The cask pool gate is removed

If the reactor building to drywell refueling bellows were to fail, which one of the following would pose the INITIAL serious radiological hazard as the cavity drains?

a. The exposed reactor vessel head studs.

b. The irradiated fuel remaining in the fuel pool.

c. Irradiated fuel remaining in the reactor vessel.

d. Irradiated components set on the floor in the cask pool.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.14	
	Importance Rating	3.4	

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 71

Proposed Answer: B

A. Incorrect - These are normally exposed before the cavity is filled and would not present a problem.

B. Correct – Of the given choices, the irradiation fuel in the fuel pool would have the highest location, so it would be the FIRST concern for radiological hazard.

C. Incorrect - The height of the reactor head flange provides sufficient shielding for the irradiated fuel in the reactor

D. Incorrect – Components stored on the floor of the cask pool are kept below the top of the irradiated fuel in the fuel pool, so they would not be the first concern for radiological hazard.

Technical Reference(s): Drawing No. BECH-M009 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank # WTS-Pilgrim 2009
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 12
55.43

(12) Radiological safety principles and procedures.

1 Point

72. With the plant at full power, the following events occur:

- A fire occurs at Bus 1A4.
- Smoke is entering the Control Room.
- The crew implements AOP 913, Fire.
- The crew selects Safe Shutdown Path CB2.

Which one of the following describes the actions required IAW AOP 913?

a. Place Control Room Ventilation in the Recirc Mode, evacuate the Control Room and establish plant control at the Remote Shutdown Panel.

b. Evacuate unnecessary personnel from the control room, don SCBAs and place Control Room Ventilation in the Fresh Air Mode.

c. Place Control Room Ventilation in the Fresh Air Mode, evacuate the Control Room and establish plant control at the Remote Shutdown Panel.

d. Evacuate unnecessary personnel from the control room, don SCBAs and place Control Room Ventilation in the Recirc Mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.27	
	Importance Rating	3.4	

Emergency procedures/Plan: Knowledge of “fire in the plant” procedures.

Proposed Question: RO Question # 72

Proposed Answer: B

A. Incorrect – with a fire at 1A4 bus, power may not be available at the RSP. Only unnecessary personnel must be evacuated from the control room and the ventilation must be placed in Fresh Air Mode

B. Correct – Per AOP 913, CB2 attachment Step1. If smoke is detected in the control room. Then –
a. Evacuate unnecessary personnel from the Control Room
b. Direct the operating crew to don SCBAs
c. Immediately halt all maintenance or surveillance testing that could cause a plant trip or transient.
Per the following NOTE: Step 2 is performed to aid in removing/preventing smoke in the Control room by going to *Fresh Air Mode*.

C. Incorrect - with a fire at 1A4 bus, power may not be available at the RSP. Evacuating the CR is not an AOP requirement

D. Incorrect – Fresh Air Mode is required.

Technical Reference(s): ACP 913 Rev 56 (CB2 attachment) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: LP 50007_94.25 Objective 94.25.01.02 (As available)

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

Question History: New Last NRC Exam: NA

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

1 Point

73. The plant was operating at full power when an immediate control room evacuation was required due to toxic gases entering the area. There is NO fire present.

IAW AOP 915, Shutdown Outside the Control, which one of the following describes action(s) required, if possible, prior to exiting the control room?

ONLY a ______

a. scram of the reactor is required.

b. scram of the reactor AND initiation of ATWS ARI/RPT are required.

c. scram of the reactor AND trip of the main turbine are required.

d. scram of the reactor, trip of the main turbine AND start of both SBDGs are required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.12	
	Importance Rating	4.0	

Emergency procedures/Plan: Knowledge of general operating crew responsibilities during emergency operations.

Proposed Question: RO Question # 73

Proposed Answer: A

A. Correct – Per AOP 915 Step 2 – If an immediate evacuation of the control room is required for a non-fire event – only a scram of the reactor is required

B. Incorrect –this would be correct for an evacuation due to a fire (AOP 915 Step 1)

C. Incorrect – A main turbine trip is not required per the AOP

D. Incorrect – A main turbine trip and start of the SBDGs is not required per the AOP

Technical Reference(s): AOP 915 Rev 39 – step 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

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

Question History: New Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 10
55.43

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

1 Point

74. Which one of the following describes conditions under which a safety limit would be violated?

   a. With reactor steam dome pressure at 1000 psig and core flow at 11% of rated, thermal power is 22% of rated thermal power.

   b. With reactor steam dome pressure at 900 psig and core flow at 9% of rated, thermal power is 22% of rated thermal power.

   c. With one recirc loop in operation, reactor steam dome pressure at 990 psig and core flow at 20% of rated, MCPR = 1.15.

   d. With two recirc loops in operation, reactor steam dome pressure at 1000 psig and core flow at 50% of rated, MCPR = 1.11.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.22	
	Importance Rating	4.0	

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: RO Question # 74

Proposed Answer: B

A. Incorrect – Per TS 2.0 – the safety limit is met

B. Correct – 2.1.1.1 Fuel Cladding Integrity – With the reactor steam dome pressure <785 psig **or** core flow < 10% rated core flow: THERMAL POWER shall be ≤ 21.7% RTP.

C. Incorrect - Per TS 2.0 – the safety limit is met

D. Incorrect - Per TS 2.0 – the safety limit is met

Technical Reference(s): TS 2.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5
	55.43	

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

1 Point

75. With the plant at full power, the following conditions exist:

- You are an NSOE coming in to start your first day of work.
- You last stood watch 96 hours ago.
- You expect to be working in the Work Control Center, unassigned to the shift.

Four hours into the shift:

One of the Control Room NSOEs is required to attend a briefing, and you have been asked to relieve him for about 1 hour.

Which one of the following identifies:

(1) how far back you are required to read the Station Log prior to taking the watch
AND
(2) whether or not the NG-016K, NSOE and ANSOE Turnover Form, must be used?

a. (1) 72 hours
(2) The Turnover form must be used.

b. (1) 72 hours
(2) The Turnover form is NOT required.

c. (1) 100 hours
(2) The Turnover form must be used.

d. (1) 100 hours
(2) The Turnover form is NOT required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.3	
	Importance Rating	3.7	

Equipment Control: Knowledge of shift or short-term relief turnover practices.
Proposed Question: RO Question # 75
Proposed Answer: A

A. **Correct.** 1st part correct, 2nd part correct. According to ACP 110.1 (p44; Rev 22), the on-coming watchstander shall review the Station Log back to the last time the individual stood the watch or three days (whichever is less). Since the operator last stood watch 96 hours ago, the requirement is to review back for ONLY 72 hours. According to ACP 1410.10 (p9; Rev 22), in a section of the procedure entitled Relieving Crew Members During the Shift, it is stated that the appropriate Turnover Form shall be used <u>any time</u> a crew member is relieved. In the event that only one position (Shift Manager, CRS, STA, NSOE, or ANSOE) is relieved, then an N/A should be placed where appropriate on the shift turnover form for the non-relieved crew members.

B. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the form does not need to be used because the relief is in the middle of a shift and of a temporary nature.

C. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because the operator last stood the watch 96 hours ago, and four hours have elapsed since the start of the work day. The operator may incorrectly believe that they are required to read the log back to the time that they last held the shift.

D. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that they are required to read the log back to the time that they last held the shift; and because the operator may incorrectly believe that the form does not need to be used because the relief is in the middle of a shift and of a temporary nature.

Technical Reference(s):	ACP 110.1 (p44; Rev 22) ACP 1410.10 (p9; Rev 22)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective:	LP 50007_1.08, Objective 1.11.01.02	(As available)

Question Source:	Bank #	DAEC Q10 1.8.1.2 P327 of RO Exam Bank	
	Modified Bank #		(Note changes or attach parent)
	New		

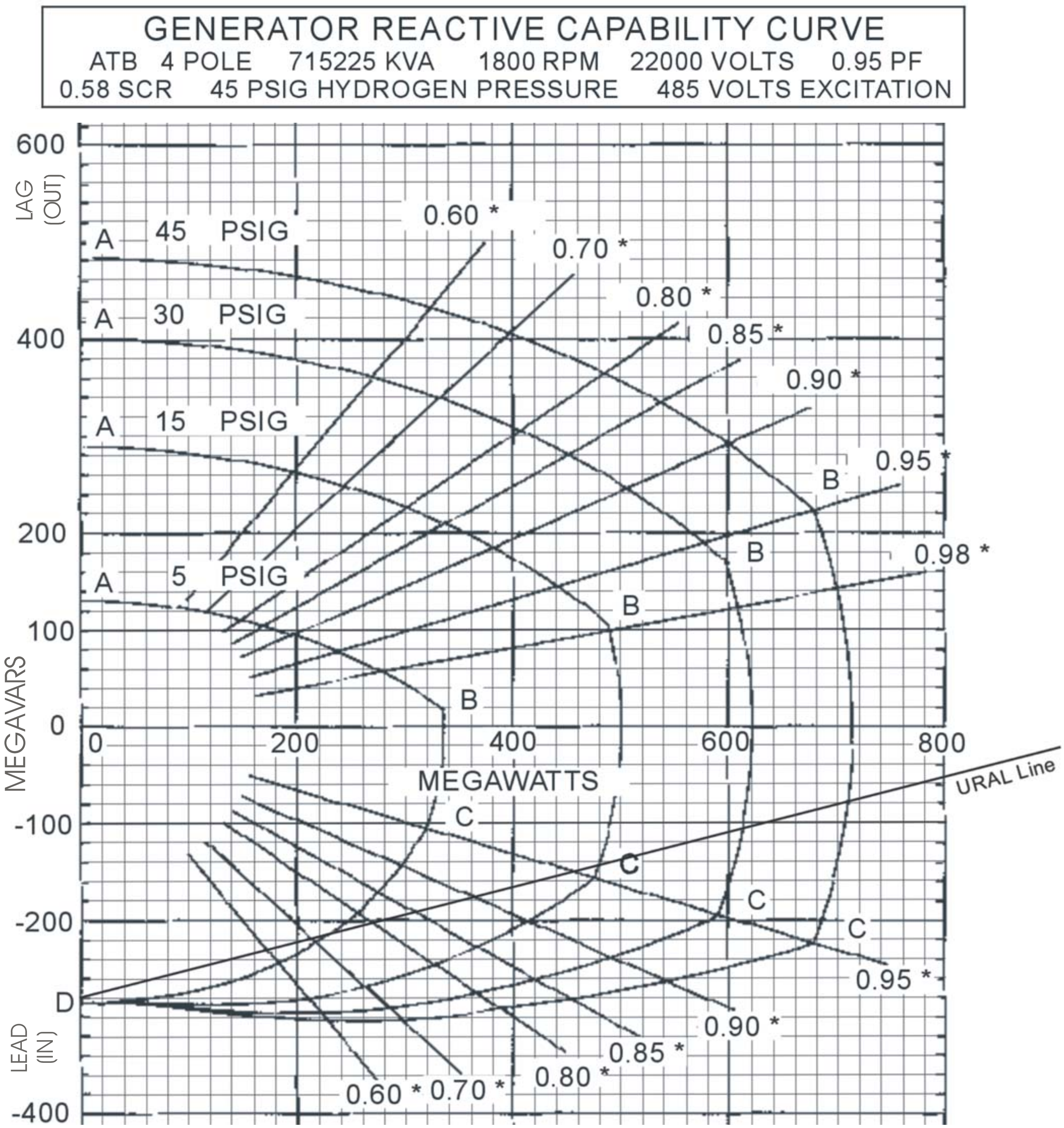
Question History: Last NRC Exam: 1999

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

APPENDIX 1
ESTIMATED CAPABILITY CURVES



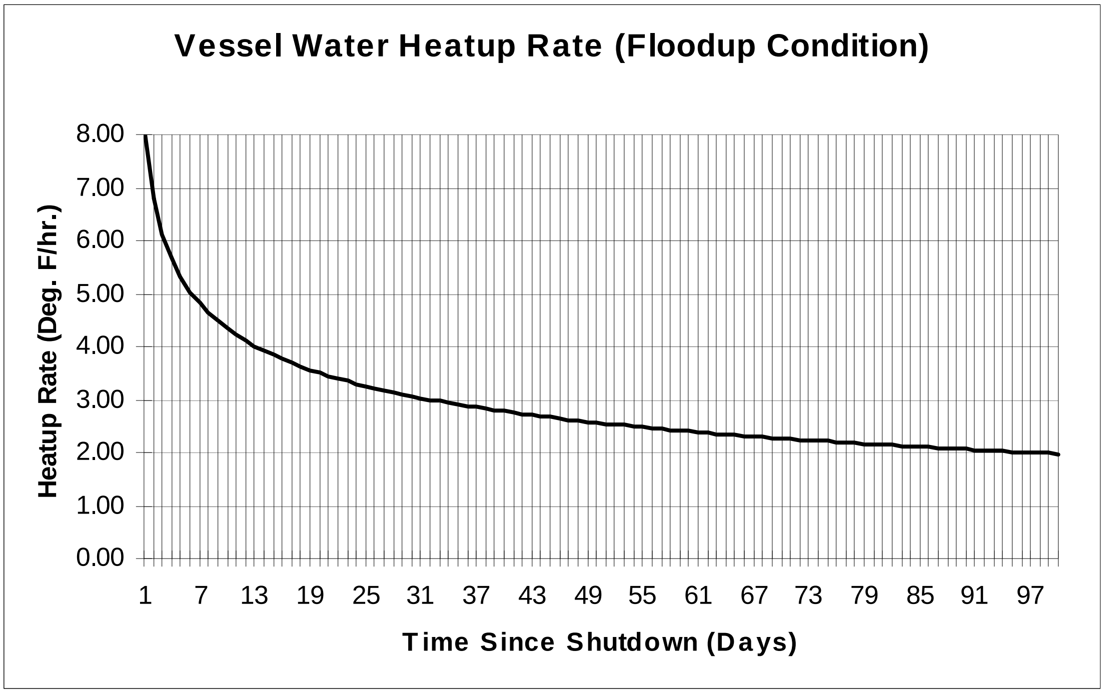
* PF (power factors)
CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

APPENDIX 1

HEATUP RATE CURVE - RPV FLOODED

NOTE

The RPV Flooded condition is defined as the RPV head removed, Spent Fuel Pool Gates removed, and the RPV and refuel cavity flooded up so that cavity level equals Spent Fuel Pool level. Spent Fuel Pool level is within the normal band.

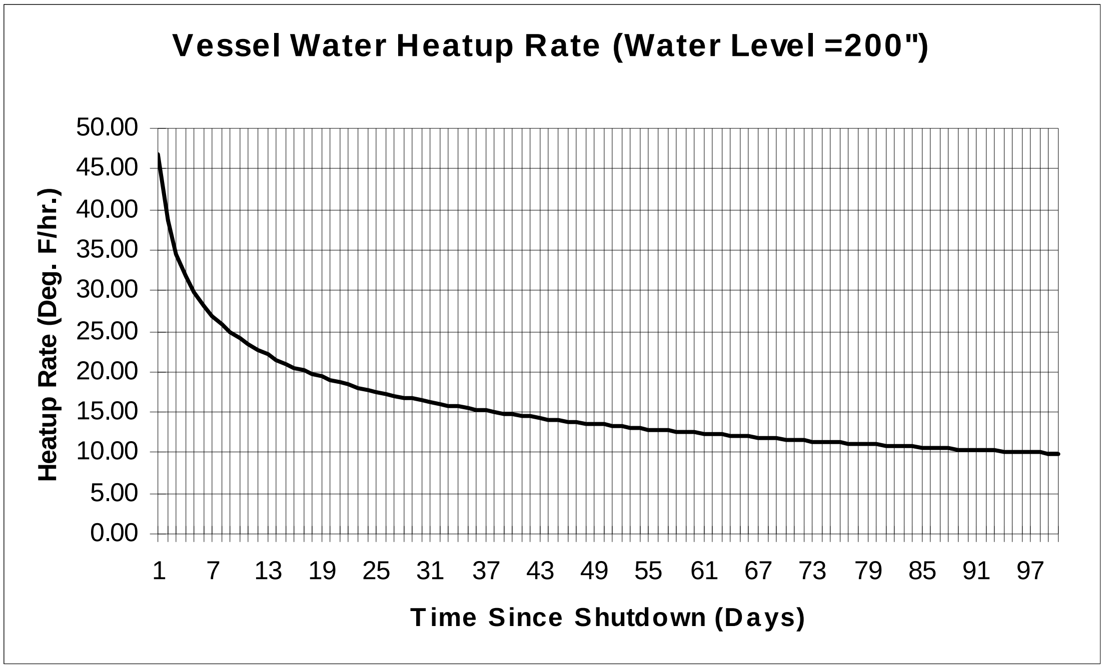


CAUTION

The initial heatup rate in the vessel may be higher than the calculated value when RHR or Fuel Pool Cooling is removed from service. The calculation used to generate the heatup rate curves assumes instantaneous mixing and heat transport from the fuel area to the remainder of the system volume. In addition, the calculated heatup rates reflect bulk temperatures not local temperatures. Under natural circulation conditions and the resulting time delay in heat transport, considerable differences in temperature may exist between the vessel and upper levels of the cavity or in the spent fuel pool. In some cases local boiling may occur but bulk boiling will not occur as long as cooling is restored within the calculated time-to-boil period.

APPENDIX 2

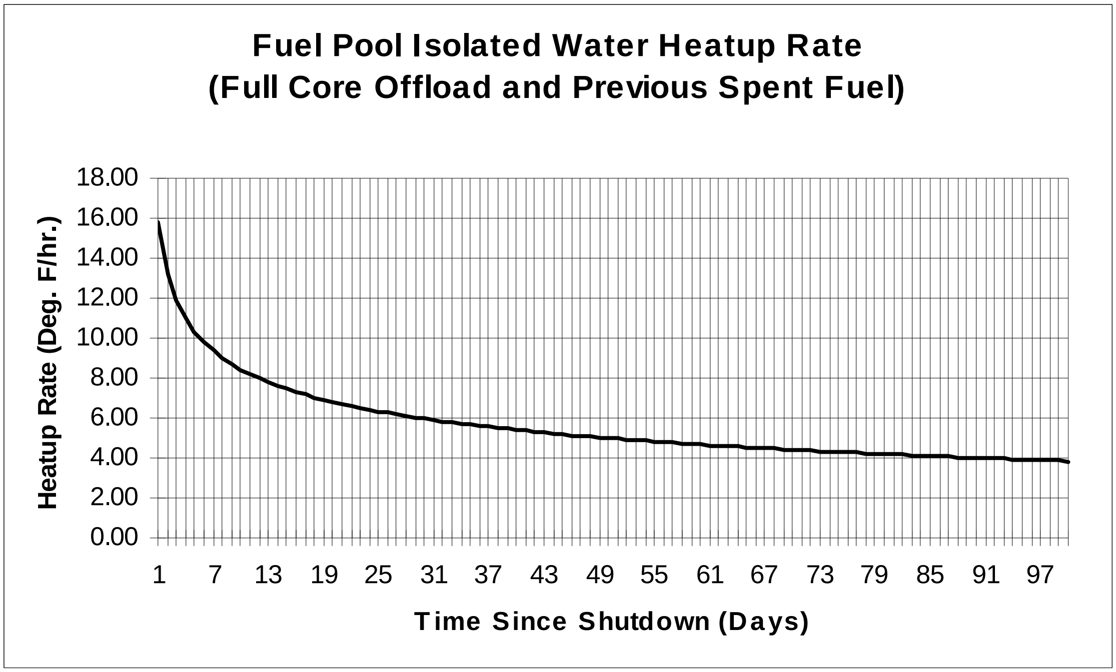
HEATUP RATE CURVE - RPV LEVEL AT 200"



CAUTION
The initial heatup rate in the vessel may be higher than the calculated value when cooling is removed from service. The calculation used to generate the heatup rate curves assumes instantaneous mixing and heat transport from the fuel area to the remainder of the system volume. In addition, the calculated heatup rates reflect bulk temperatures not local temperatures. Under natural circulation conditions and the resulting time delay in heat transport, considerable differences in temperature may exist between the fuel area and upper levels of vessel. In some cases local boiling may occur but bulk boiling will not occur as long as cooling is restored within the calculated time-to-boil period.

APPENDIX 3

LOSS OF FUEL POOL COOLING HEATUP RATE CURVE



CAUTION

The initial heatup rate in the spent fuel pool may be higher than the calculated value when cooling is removed from service. The calculation used to generate the heatup rate curves assumes instantaneous mixing and heat transport from the fuel area to the remainder of the system volume. In addition, the calculated heatup rates reflect bulk temperatures not local temperatures. Under natural circulation conditions and the resulting time delay in heat transport, considerable differences in temperature may exist between the fuel area and measured temperatures in fuel pool cooling heat exchanger inlets.

	WRITTEN/ORAL EXAMINATION COVERSHEET

Trainee Name:			
Employee Number:		**Site:**	DAEC
Examination Number/Title: *Series A, Rev. 0, 2009 NRC Senior Reactor Operator Initial License Exam*			
Training Program: *Initial License Training*			
Course/Lesson Plan Number(s): *50007 / Various*			
Total Points Possible: 25	**PASS CRITERIA:** ≥ **70%**		**Grade: __/**25 **= __%**
Graded by:			**Date:**
Co-graded by (if necessary):			**Date:**

EXAMINATION RULES

1.	References may not be used during this examination, unless otherwise stated.
2.	Read each question carefully before answering. If you have any questions or need clarification during the examination, contact the examination proctor.
3.	Conversation with other trainees during the examination is prohibited.
4.	Partial credit will not be considered, unless otherwise stated. Show **all** work and state **all** assumptions when partial credit may be given.
5.	Rest room trips are limited and only one examinee at a time may leave.
6.	For exams with time limits, you have **120 (2 Hours)** minutes to complete the examination.
7.	Feedback on this exam may be documented on QF-1040-13, Exam Feedback Form. Contact Instructor to obtain a copy of the form.

EXAMINATION INTEGRITY STATEMENT

Cheating or compromising the exam will result in disciplinary actions up to and including termination.

"I acknowledge that I am aware of the Examination Rules stated above. Further, I have not given, received, or observed any aid or information regarding this examination prior to or during its administration that could compromise this examination."

Examinee's Signature: **Date:**

REVIEW ACKNOWLEDGEMENT

"I acknowledge that the correct answers to the exam questions were indicated to me following the completion of the exam. I have had the opportunity to review the examination questions with the instructor to ensure my understanding.

Examinee's Signature: **Date:**

Retention: 6 years
Retain in: Training Records

1. During an accident the following plant conditions exist:

   - RPV pressure 600 psig
   - RPV water level +100 inches
   - Drywell pressure 19 psig
   - Torus water level 7.5 ft
   - Torus pressure 18 psig

   Which one of the following is required based upon the above conditions?

   a. Enter EOP-ED and emergency depressurize using the ADS SRVs.

   b. Anticipate ED and rapidly depressurize with the bypass valves.

   c. IAW EOP-1, RPV Control, cycle SRVs in sequence to establish a reactor cooldown at a rate <100°F/hour.

   d. IAW EOP-1, RPV Control, cool down the RPV with the main turbine bypass valves or Alternate Pressure Control Systems (Table 7).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295030	EA2.01
	Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool level

Proposed Question: SRO Question # 76

Proposed Answer: A

A. Correct –UNSAFE PSPL due to combination of low suppression pool level and high suppression chamber pressure EOP-02-PCC requires emergency depressurization. With Torus Water level above 4.5 feet ADS SRVs are used.

B. Incorrect – ED is required at this point and with Torus Water level above 4.5 feet ADS SRVs are used.

C. Incorrect – Must ED per procedure and OPEN 4 ADS SRVs.

D. Incorrect – Torus Water level is low but not low enough to require alternate emergency depressurization.

Technical Reference(s): EOP-2, Step PC/P-7
PSPL Curve
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-2, T/L & PC/P legs
PSPL Curve

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

2. While at 100% power, a partial loss of 125 VDC has rendered the 1D14 bus de-energized.

   Which one of the following describes an effect on RCIC and an associated Technical Specification requirement?

   a. The RCIC steam supply inboard isolation valve MO-2400 has lost power. Immediately verify by administrative means that HPCI is operable.

   b. The RCIC steam supply outboard isolation valve MO-2401 has lost power. Immediately verify by administrative means that HPCI is operable.

   c. The RCIC steam supply inboard isolation valve MO-2400 has lost power. Immediately run HPCI to verify operability.

   d. The RCIC steam supply outboard isolation valve MO-2401 has lost power. Immediately run HPCI to verify operability.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295004	AA2.04
	Importance Rating		3.3

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : System lineups

Proposed Question: SRO Question # 77

Proposed Answer: B

A. Incorrect - The 1D14 bus affects the RCIC outboard isolation valve IAW SD 959.1

B. Correct – IAW TS 3.5.3 –The 1D14 bus affects the RCIC outboard isolation valve IAW SD 959.1. RCIC MO-2404 also loses power and this valve is normally closed. Therefore RCIC would be inoperable with this power loss. This LCO requires that HPCI be verified Operable by Admin means Immediately, not run it..

C. Incorrect –The power supply issue affects the outboard valve. This LCO requires that HPCI be verified Operable by Admin means Immediately.

D. Incorrect - This LCO requires that HPCI be verified Operable by Admin means Immediately, not run it.

Technical Reference(s): T.S. 3.5.3 Condition A
AOP 302.1, page 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source:	Bank #	DAEC SRO Bank, Ques 2, pg 166	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

1 Point

3. Following a spurious Main Turbine Trip and an ATWS, the following conditions exist:

   - RPV water level was lowered reducing reactor power.
   - All APRMs indicate downscale
   - RPV water level has been restored and is at +190”
   - SBLC has been injecting and tank level has reached 14%
   - A majority of control rods remain stuck out of the core

   Which one of the following actions is required at this time?

   a. Exit ATWS RPV Control and enter EOP 1, RPV Control.

   b. Cool down and place Shutdown Cooling in service using SEP-306, Initiation of SDC for EOP Use.

   c. Terminate boron injection and maintain RPV water level to 170” to 211” IAW EOP 1, RPV Control.

   d. Maintain RPV water level using a Core Spray Pump IAW OI-151, Core Spray System.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295037	EA2.03
	Importance Rating		4.4

Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC Tank Level.

Proposed Question: SRO Question # 78

Proposed Answer: B

A. Incorrect – The criteria to exit ATWS-RPV Control is not met, ie all rods are not inserted and/or RE has not determined the reactor will remain shutdown under all conditions without boron.

B. Correct – With Cold Shutdown Boron Weight injected the reactor may be cooled down and shutdown cooling placed in service.

C. Incorrect – There is no direction to terminate injection. Injection should continue until the full contents of the SBLC tank are injected.

D. Incorrect – RPV water level can be restored at Hot Shutdown Boron Weight. However restoring water level is done with preferred systems and Core spray is not a preferred system.

Technical Reference(s): ATWS-RPV Control, /P-5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: ATWS RPV Control /L without setpoints

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

4. The plant was operating at full power.

   The control room must be evacuated due to a fire. All required control room actions were completed prior to the evacuation

   Which one of the following describes a task that must be completed IAW AOP 915, Shutdown Outside the Control Room?

   a. To attempt closure of a spuriously opened SRV, transfer control to Remote Shutdown Panel 1C388 ONLY.

   b. To attempt closure of a spuriously opened SRV, transfer control to Remote Shutdown Panels 1C388 and 1C389.

   c. Establish additional ventilation in the 1A3 and 1A4 switchgear rooms within 1 hour.

   d. Establish additional ventilation in the 1A3 switchgear room ONLY within 1 hour.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	2.4.35
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects. (Reactor Low Water Level)

Proposed Question: SRO Question # 79

Proposed Answer: B

Explanation (Optional):

A. Incorrect. IAW AOP 915 – Transfer of panel 1C389 is also required for SRV control

B. Correct. IAW AOP 915, Per caution on Page 6 – "For Control Room evacuation as the result of a fire, transfer of control at panels 1C388, 1C389, 1C390, 1C391, 1C392 is required to be completed within 20 minutes".

C. Incorrect. the requirement has no time constraints

D. Incorrect. the requirement has no time constraints

Technical Reference(s): AOP-915 Rev 39 / ACP 103.10 Rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

5. The Main Turbine Generator has just been placed on the grid and is operating at 300 MWe during a reactor startup.

02:35 Annunciator 1C07B (B-2) TURBINE HI VIBRATION alarms.

02:36 The crew identifies that the Main Turbine bearing #8 vibrations are rising slowly per the PPC and the Turbine Vibration Expansion Recorder VR-9019.

02:37 The CRS established Turbine Vibrations as a critical parameter.

02:43 Annunciator 1C07B (A-2) TURBINE HI-HI VIBRATION alarms.

02:43 Bearing #8 vibrations indicate 10 mils and continue to rise slowly.

03:00 The crew identifies bearing #7 vibrations are 6 mils and rising slowly; bearing #8 is now 10.7 mils. The crew commences lowering power IAW ARP 1C07B (A-2).

03:03 Bearing #8 vibrations indicate 12.2 mils and bearing #7 is now 10.1 mils with both continuing to rise slowly.

Which action(s) would you direct based on these conditions?

a. Perform a fast power reduction per IPOI-4 and continue to monitor bearing vibrations.

b. Continue the normal power reduction per IPOI-3 and continue to monitor bearing vibrations.

c. Perform a fast power reduction per IPOI-4, scram the reactor and trip the turbine.

d. Continue the normal power reduction per IPOI-3, scram the reactor and trip the turbine when vibration reaches 15 mils.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295005 AA2.02	
	Importance Rating		2.7

295005 Main Turbine Generator Trip
Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : AA2.02 Turbine vibration (CFR: 41.10 / 43.5 / 45.13)
Proposed Question: SRO Question # 80

Proposed Answer: C

A. Incorrect –AOP 693 directs fast power reduction, reactor scram, Turbine Trip, shutting MSIVs, and breaking vacuum when Bearing #8 exceeds 12 mils for given conditions (one bearing exceeding 12 mils with another bearing trending up.)
B. Incorrect –ARP 1C07A A-2 directs entry into AOP 693 if bearing vibrations exceeds 12 mils.
C. Correct –AOP 693 directs fast power reduction, reactor scram, Turbine Trip, shutting MSIVs, and breaking vacuum when Bearing #8 exceeds 12 mils for given conditions (one bearing exceeding 12 mils with another bearing trending up.)
D. Incorrect –ARP 1C07A A-2 directs entry into AOP 693 if bearing vibrations exceeds 12 mils.

Technical Reference(s): ARP 1C07B A-2
AOP 693, section 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.43 5
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

6. A Group 1 isolation and small break LOCA has occurred and the following conditions exist:

• All control rods	FULL IN
• RPV pressure	Controlling on LO-LO Set
• RPV level	155", rising slowly
• Torus level	11 feet, stable
• Torus pressure	12 psig, rising slowly
• Drywell temperature	220°F, rising slowly

The operators attempted to place Torus Cooling in service but were not successful.

The STA reports that SPDS torus water temperature is reading 155°F and Graph 4, Heat Capacity Limit, limits are being approached.

Which one of the following actions is required for these conditions?

a. Immediately lower reactor pressure with SRVs based on SPDS Graph 4, Heat Capacity Limits, trend.

b. After verifying computer points are not marked with a YELLOW “V”, lower reactor pressure with SRVs based on SPDS Graph 4, Heat Capacity Limits, trend.

c. Confirm the SPDS reading by checking the 1C03 panel indications and, only if validated, exit EOP-2, Primary Containment Control and enter EOP-ED and emergency depressurize.

d. Confirm the SPDS readings by checking the 1C03 panel indications and, only if validated, lower reactor pressure with SRVs based on the EOP 2 Graph 4, Heat Capacity Limits, plot.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025	2.1.19
	Importance Rating		3.8

Conduct of Operations: Ability to use plant computers to evaluate system or component status. (High Reactor Pressure)

Proposed Question: SRO Question # 81

Proposed Answer: D

A. Incorrect. IAW OI-831.4, No Emergency action should be taken based on the SPDS data alone.

B. Incorrect. IAW OI-831.4, No Emergency action should be taken based on the SPDS data alone.

C. Incorrect. There is no requirement or need to exit EOP-2 and ED.

D. Correct. This value of torus temperature / reactor pressure requires a lowering of reactor pressure to maintain it within the safe region of the curve. SPDS data must be confirmed with panel indications prior to taking actions

Technical Reference(s): OI-831.4, Rev 64, Sect. 6, caution pg 35
EOP-2, step T/T-6 and HCTL curve
SD-831.4a, page 51 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Torus Temp leg of EOP-2 and HCTL Curve

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2005

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

7. The plant is operating at full power.

   The control room receives a call from ITC Midwest stating that lightning strikes have led to a degraded grid condition and a contingency trip of Duane Arnold would lead to an undervoltage condition in the DAEC switchyard 161 KV bus.

   15 minutes after the ITC Midwest call, annunciator 1C-08C (B-4), MAIN GENERATOR FIELD MAX EXCITATION, alarms. 10 seconds later the alarm has not cleared.

   Which one of the following describes:

   (1) action(s) required due to the notification from ITC Midwest

   AND

   (2) action(s) required due to the alarm condition?

   a. (1) Declare both Offsite Sources Inoperable IAW Technical Specifications.
      (2) Shift to manual voltage control IAW AOP 304, Grid Instability.

   b. (1) Declare both Offsite Sources Inoperable IAW Technical Specifications .
      (2) Verify the main generator has tripped and enter IPOI-5, Reactor Scram.

   c. (1) Start and load both SBDGs IAW OI 304.2, 4160V/480V Essential Electrical Distribution System.
      (2) Shift to manual voltage control IAW AOP 304, Grid Instability.

   d. (1) Start and load both SBDGs IAW OI 304.2, 4160V/480V Essential Electrical Distribution System.
      (2) Verify the main generator has tripped and enter IPOI-5, Reactor Scram.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	700000	AA2.08
	Importance Rating		4.3

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Criteria to trip the turbine or reactor

Proposed Question: SRO Question # 82

Proposed Answer: B

A. Incorrect – IAW AOP 304 - The AUTO Voltage Regulator will maintain generator operation within the generator capability curve. Operation of the over excitation limiter initiates annunciator 1C08C B-4. Once the limiter is initiated the auto voltage regulator may be limiting excitation of the generator. Shifting to Manual Voltage Control under these conditions may cause a generator trip. Because a trip would have already occurred, this action is not correct.

B. Correct – IAW AOP 304 - **IF** notified by ITC Midwest that the contingency of a trip of the DAEC would lead to an undervoltage condition of < 99.2% in the DAEC switchyard 161 KV bus, **THEN** Declare both Offsite Sources inoperable and enter TS LCO actions as required by the mode of applicability. IAW ARP 1C-08C (B-4) - If the overvoltage condition exists for longer than 5 seconds, the Voltage Regulator transfers from AUTOMATIC to MANUAL. The following occurs; *If either or both generator output breakers are closed, the generator trip will be via the Generator Backup Lockout Relay 286/B.* With the plant online both generator output breakers are closed, the generator will trip. If the generator trips and power is above 26%, a reactor scram and entry to IPOI 5 is required.

C. Incorrect – Per AOP 304 CAUTION - It is not appropriate to manually start and load a SBDG during degraged grid condtions. Do not use OI 304.2, section 7.6 TRANSFERRING ESSENTIAL BUS 1A3[4] FROM STARTUP OR STANDBY TRANSFORMER TO STANDBY DIESEL GENERATOR to attempt to put the essential buses on the SBDGs without the approval of Operations Management. Shifting to Manual Voltage Control under these conditions may cause a generator trip. Because a trip would have already occurred, this action is not correct.

D. Incorrect - Per AOP 304 CAUTION - It is not appropriate to manually start and load a SBDG during degraged grid condtions. Do not use OI 304.2, section 7.6 TRANSFERRING ESSENTIAL BUS 1A3[4] FROM STARTUP OR STANDBY TRANSFORMER TO STANDBY DIESEL GENERATOR to attempt to put the essential buses on the SBDGs without the approval of Operations Management.

Technical Reference(s): ARP 1C08C, (B-4) Rev 46
AOP-304, Rev 22
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2,5

(2) Facility operating limitations in the technical specifications and their bases.
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

8. A reactor scram has occurred from full power due to a complete Loss of Uninterruptible AC power. All 8 RPS Scram white lights are extinguished, but the 1C05 operator cannot confirm that all rods are fully inserted.

   All LPRM downscale lights are on and when the IRMs are fully inserted, they read between range 3 and 4 and are lowering.

   RPV pressure is ≈900 psig and rising very slowly with the Main Steam Line Drains open. SBLC was not injected.

   (1) Is the reactor considered SHUTDOWN UNDER ALL CONDITIONS WITHOUT BORON?

   AND

   (2) How is the ATWS EOP utilized in this situation?

   a. (1) NO
      (2) Exit the ATWS EOP and perform IPOI-5.

   b. (1) NO
      (2) Exit only the /Q leg of the ATWS EOP.

   c. (1) YES
      (2) Exit the ATWS EOP and perform IPOI-5.

   d. (1) YES
      (2) Exit only the /Q leg of the ATWS EOP.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295015	AA2.01
	Importance Rating		4.3

Ability to determine and / or interpret the following as they apply to INCOMPLETE SCRAM: Reactor power

Proposed Question: SRO Question # 83

Proposed Answer: B

A: Incorrect – Only the q leg of the ATWS EOP may be exited. The entire EOP may not be exited until it is determined that you are shutdown under all conditions

B: Correct – Per ATWS EOP Bases Discussion Page 4, "Shutdown under ALL conditions without boron" can be determined by relying on the Technical Specification demonstration of adequate shutdown margin:
• One control rod is out beyond position 00
• All other control rods are at position 00
For other combinations of rod patterns and boron concentration, reactor engineering will need to perform a shutdown margin calculation.

When either of the conditions identified in the Continuous Recheck Statement is achieved, it is appropriate to terminate boron injection, exit the ATWS EOP, and enter EOP 1 for control of the transient.
Since these conditions are not given, the EOP may not be exited.

C: Incorrect – The conditions stated in the question stem do not meet the EOP Bases definition of Shutdown under ALL conditions without boron"

D: Incorrect - The conditions stated in the question stem do not meet the EOP Bases definition of Shutdown under ALL conditions without boron". The entire EOP would exited if that were the case.

Technical Reference(s): EOP ATWS bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X - 21075
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 5, 6

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

9. An unisolable coolant system leak has occurred in the Reactor Building that has resulted in RPV level lowering to 17”.

   Operators recovered RPV level and were attempting to stabilize the plant when they noticed the following:

   - A RED annunciator on panel 1C-35A (C-3) for REACTOR BLDG KAMAN 3, 4, 5 ,6, 7, & 8 HI RAD OR MONITOR TROUBLE
   - PPC indicates that a Reactor Building Kaman Hi-Hi alarm exists

   The Kaman readings are as follows:

   - REACTOR BLDG KAMAN 5/6 concentration is 9.3 E-3 uCi/cc
   - REACTOR BLDG KAMAN 7/8 concentration is 6.3 E-2 uCi/cc

   The Reactor Building Exhaust Fans (1V-EF-11A & B) and the Main Plant Exhaust Fans (1V-EF-1, 2, & 3) responded as designed.

   What actions must be directed and what Emergency Action Level must be declared?

   a. Direct operators to TRIP the Main Plant Exhaust Fans.
      If the above REACTOR BLDG KAMAN readings continue for 15 minutes, offsite Rad Conditions will then exceed the Site Area Emergency level.
      Because RPV lowered to 17” before recovering, an Alert must be declared.

   b. Direct operators to RESTART the Reactor Building Exhaust Fans.
      If the above REACTOR BLDG KAMAN readings are expected to continue for 15 minutes, offsite Rad Conditions will exceed the Site Area Emergency level.
      Because RPV lowered to 17” before recovering, a Site Area Emergency must be declared.

   c. Direct operators to TRIP the Main Plant Exhaust Fans.
      With the above REACTOR BLDG KAMAN readings, a Site Area Emergency must be declared.

   d. Direct operators to RESTART the Reactor Building Exhaust Fans.
      With the above REACTOR BLDG KAMAN readings, an Alert must be declared.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295017	2.4.41
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications. (High offsite release rate)

Proposed Question: SRO Question # 84

Proposed Answer: C

A: Incorrect – The KAMAN levels have already exceeded the SAE criteria. The 15 minutes is associated with the Alert classification. There is no SAE classification for RPV level at 17”.

B: Incorrect – Selected if the RB Kaman monitors are believed to be in the RB Vent Shaft rather than on the discharge of the MP Exhaust Fans. Operators are directed to restart the Turbine Bldg Exhaust Fans, not Reactor Building Exhaust Fans. There is no SAE classification for RPV level at 17”.

C: Correct - At <170 inches a Group 3 isolation occurs which trips EF-11A&B, closes 1V-EF-13A & B, and aligns SBGT to draw on the RB Vent Shaft. EF1/2/3 continue to run and draw on the Main Plant Exhaust Plenum. The RB Vent Shaft and the MP Exhaust Plenum are physically separated by only a wall which, in the history of the plant, has been found to be cracked. Also the dampers 1V-EF-13A/B could be leaking, also allowing the RB Vent Shaft to flow to the MP Exhaust Plenum and out past 1V-EF-1/2/3 which normally continue to run after a Group 3 isolation. This is a real enough concern that there is a P&L in the Reactor Building HVAC OI, a Continuous Recheck statement in EOP-4 and Steps in ARP 1C35A C-3 step 3.3.a.
Per EAL Bases Document EBD-R Table on page 5, the SAE Level is exceeded REACTOR BLDG KAMAN 7/8 release rate.

D: Incorrect - Selected if the RB Kaman monitors are believed to be in the RB Vent Shaft rather than on the discharge of the MP Exhaust Fans. Operators are directed to restart the Turbine Bldg Exhaust Fans, not Reactor Building Exhaust Fans. The KAMAN levels have already exceeded the SAE criteria

Technical Reference(s): EBD-R page 5 table (EAL bases)
ARP 1C35A C-3. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL Matrix

Learning Objective: (As available)

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

Original Question:

An unisolable coolant system leak has occurred in the Reactor Building that has resulted in RPV level lowering to the point that fuel became uncovered and fuel damage occurred.

Operators recovered RPV level and were attempting to stabilize the plant when they noticed a RED annunciator on panel 1C35 for REACTOR BLDG KAMAN 3, 4, 5 ,6 , 7,& 8 HI RAD OR MONITOR TROUBLE.

The Reactor Building Exhaust Fans (1V-EF-11A & B) and the Main Plant Exhaust Fans (1V-EF-1, 2, & 3) responded as designed.

What could be the cause of this alarm and what actions must be directed regarding these fans to mitigate this condition?

a. The Main Plant Exhaust Fans must still be drawing on the Reactor Building Vent Shaft. Direct operators to TRIP the Main Plant Exhaust Fans.

b. The Main Plant Exhaust Fans will have tripped causing a high concentration of activity at the monitors. Direct operators to RESTART the Main Plant Exhaust Fans.

c. The Reactor Building Exhaust Fans must still be drawing on the Reactor Building Vent Shaft.
Direct operators to TRIP the Reactor Building Exhaust Fans.

d. The Reactor Building Exhaust Fans will have tripped causing a high concentration of activity at the monitors. Direct operators to RESTART the Reactor Building Exhaust Fans.

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

(1) Conditions and limitations in the facility license.

1 Point

10. A Loss of Coolant Accident has occurred which requires operators to perform SEP 301.1, Torus Vent via SBGT. The following conditions exist:

- One train of Standby Gas Treatment (SBGT) is in operation
- Drywell pressure is 50 psig and still rising slowly
- Three Torus vent valves are open
  - o CV-4301, OUTBD TORUS VENT ISOL.
  - o CV-4309, INBD TORUS VENT BYPASS ISOL.
  - o CV-4300, INBD TORUS VENT ISOL.

After opening CV-4300, airborne activity and radiation levels on Reactor Building Second Floor (El. 786 ft.) have risen dramatically.

Which of the following has caused this condition and what actions are required to continue venting?

a. The SBGT inlet relief damper has opened due to excessive pressure; start the standby SBGT Train IAW OI 170, SBGT System, to raise SBGT system flow.

b. The SBGT inlet relief damper has opened due to excessive pressure; assess the need for venting and use the Hard Pipe Vent per SEP 301.3 as required.

c. The Hard Pipe Vent rupture disc has ruptured; assess the need for venting and shift to Drywell vent per SEP 301.2 as required.

d. The Hard Pipe Vent rupture disc has ruptured; throttle MO-4309A, BYPASS VENT THROTTLE, as needed to lower pressure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295033	EA2.03
	Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Cause of high area radiation

Proposed Question: SRO Question # 85

Proposed Answer: B

A. Incorrect – this provides no additional flow and does not lower pressure

B. Correct – Per SEP 301.1, If SBGT inlet pressure approaches 10" WG, assess the need for continued venting and/or use of the Hard Pipe Vent per SEP 301.3. Caution, If SBGT inlet pressure exceeds 10" WG, the SBGT inlet relief damper will open and relieve pressure into the RB 786' Level.

C. Incorrect – The hard pipe vent rupture disc does not discharge inside the Reactor Building.

D. Incorrect – Throttling with MO-4309A is specifically prohibited by SEP 301.1 CAUTION, it has a non-essential power supply and may impede venting.

Technical Reference(s): SEP 301.1, Rev 6 Step 7 and caution pg 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 4, 5

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

11. The plant is at full power. All systems are operable.

    Then, annunciator 1C-03A (C-8), "A" CORE SPRAY SPARGER LO ∆P, alarms. The operators confirm it is a valid alarm.

    Based on the above condition an "A" Core Spray piping leak/break has occurred __(1)__. An immediate shutdown __(2)__ required.

    a. (1) INSIDE the Core Shroud
       (2) is

    b. (1) INSIDE the Core Shroud
       (2) is NOT

    c. (1) BETWEEN the Reactor Pressure Vessel wall and the Core Shroud
       (2) is.

    d. (1) BETWEEN the Reactor Pressure Vessel wall and the Core Shroud
       (2) is NOT.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	209001	A2.05
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Core Spray line break

Proposed Question: SRO Question # 86

Proposed Answer: D

A:	Incorrect – The alarm is not an indication of an inside the shroud break based upon its tap off point on the LPCS piping. Entry conditions for LCO 3.0.3 have not been met with only one core spray loop inoperable therefore an immediate shutdown is not required
B:	Incorrect - The alarm is not an indication of an inside the shroud break based upon its tap off point on the LPCS piping.
C:	Incorrect - Entry conditions for LCO 3.0.3 have not been met with only one core spray loop inoperable therefore an immediate shutdown is not required.
D:	Correct – Per ARP 1C-03A (C-8) – this alarm is from the Core Spray System Header to top of the Core plate and caused by differential pressure low. Entry conditions for LCO 3.0.3 have not been met with only one core spray loop inoperable therefore an immediate shutdown is not required.

Technical Reference(s): ARP 1C03A (C-8) Rev 48
TS 3.5.1.B
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

12. The plant is in HOT SHUTDOWN. The “B” Shutdown Cooling (SDC) Loop is in service with “B” RHR and “B” RHRSW pumps running.

MO1940, RHR HX 1E-201B BYPASS, and MO1939, RHR HX 1E-201B INLET THROTTLE, valves are THROTTLED in mid position.

- MO1904 and MO1905, RHR Loop “B” Inject Isolation Valves are OPEN.
- MO1908 and MO1909, RHR Shutdown Cooling Suction Isolation Valves are OPEN.

Annunciator 1C03B (B-4), RHR SHUTDOWN COOLING SUCTION HEADER HI PRESSURE, alarms and SDC Header pressure is reported to be 105 psig and rising at 2 psig per minute.

You direct the operators to raise the cooldown rate.

Several minutes later, the 1C03 operator reports that MO1940 would not move from its throttled position, RHR “B” flow has been maximized through the heat exchanger, and RHR suction header pressure is at 140 psig.

No other plant conditions have changed.

Based on these plant conditions, you direct the operators to ________

a. start the “D” RHRSW pump and raise RHRSW flow IAW OI 416, RHRSW System. Enter the Technical Specification Limiting Condition for Operation for LPCI.

b. throttle OPEN more on MO1939 and start the “D” RHR pump if necessary. Enter the Technical Specification Limiting Condition for Operation for LPCI.

c. CLOSE MO1905, TRIP the “B” RHR pump, and CLOSE MO1908 and MO1909. Enter AOP 149, Loss of Decay Heat Removal.

d. CLOSE MO1939, Start the “D” RHR pump and then re-establish SDC flow. Enter AOP 149, Loss of Decay Heat Removal, until SDC is re-established.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	223002	A2.03
	Importance Rating		3.3

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: System logic failures

Proposed Question: SRO Question #87

Proposed Answer: C

A: Incorrect – SDC should have isolated at 135 psig. The ARP for a Group 4 should be carried out. Increasing RHRSW flow is not part of the ARP guidance

B: Incorrect - ARP 1C03B B-4 directs increasing cooldown with MO 1939 and another pump would help flow. T.S. should be entered on failure of MO-1940. However, the plant is above the PCIS Group 4 pressure and SDC should be promptly removed and isolated.

C: Correct - The initial alarm indicates an increase in RPV temperature and pressure. The ARP directs increasing the cooldown rate to lower pressure, which was directed. At 135 psig a PCIS group 4 should have occurred but did not. ARP 1C05B D-8 "PCIS Group 4 Isolation" should be in alarm and SDC secured. The CRS should direct the actions from the ARP that did not occur. In this case securing SDC is appropriate. Also entry into AOP 149 is directed.

D: Incorrect - Starting a second RHR pump would increase flow. AOP 149 entry is correct when SDC is lost and recovery of SDC will be the goal. However, the plant is above the PCIS Group 4 pressure("D" RHR pump won't start under these conditions) and SDC should be promptly removed and isolated as directed in ARP 1C05B D-8 for pressure protection of the RHR piping.

Technical Reference(s): 1C03B B-4 Rev 36
1C05B D-8 Rev 81 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2002

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

13. A plant startup is in progress and the Mode Switch is ready to be placed in RUN.

The only inoperable equipment is IRM “B” which is bypassed due to a downscale failure. I&C work is in progress.

Then, a half scram and a Rod Block occurs on RPS Channel “B”.

I&C reports they lifted the wrong lead in the IRM panels and caused an INOP trip on IRM “D”.

Which one of the following describes whether the Technical Specification (TS) actions have been met and whether TS permits the Mode Switch to be taken to RUN in this condition?

a. The TS required actions are already met with the trip on RPS Channel “B”.
   The Mode Switch may NOT be taken to RUN until at least one of the IRMs (“B” or “D”) is declared operable.

b. The TS required actions are already met with the trip on RPS Channel “B”.
   The Mode Switch may be taken to RUN because the IRM TS does not apply in MODE 1.

c. The TS required actions are NOT met.
   The Mode Switch may NOT be taken to RUN until at least one of the IRMs (“B” or “D”) is declared operable.

d. The TS required actions are NOT met.
   The Mode Switch may be taken to RUN because the IRM TS does not apply in MODE 1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	215003	2.2.36
	Importance Rating		4.2

Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: SRO Question #88

Proposed Answer: B

A: Incorrect – TS 3.3.1.1.A requires the channel to be in the tripped condition within 12 hours. This is met with the RPS trip. Since the IRMs are not required in mode 1, the mode switch may be moved.

B: Correct - TS 3.3.1.1.A requires the channel to be in the tripped condition within 12 hours. This is met with the RPS trip. TS 3.0.4 permits a mode change to a mode where the TS does not apply if a risk assessment and establishment of risk management actions is performed first.

C: Incorrect – The TS actions are met and the mode switch may be moved.

D: Incorrect - TS 3.3.1.1.A requires the channel to be in the tripped condition within 12 hours. This is met with the RPS trip.

Technical Reference(s): TS 3.3.1.1
TS 3.0.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO RPS instrumentation Tables
No TS Section 3.0

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

1 Point

14. The plant is currently in an electrical ATWS with the following conditions:

- ADS is locked out
- The MSIVs are Closed
- Defeat 11, Containment N2 Supply Isolation Defeat, has been installed
- Reactor power is cycling between 25% and 55% power
- RPV level is 80 inches and being intentionally lowered
- SBLC is injecting
- The RIPs are being implemented
- RPV Pressure is being maintained automatically between 1080 psig and 1130 psig.

Which one of the following describes a required action, if any, based on the above conditions?

a. The opening and closing SRV may cause significant power transients but all systems are operating as designed, so NO additional EOP actions are required.

b. The main concern in this condition is that an SRV could stick open.
Place HPCI in service IAW OI 152 QRC 1, HPCI Rapid Start, and/or RCIC in service IAW OI 150 QRC 1, RCIC Rapid Start, in CST to CST mode for pressure control.

c. The opening and closing of the SRVs exerts significant dynamic loads on the SRV tailpipes and support structures. Manually open SRVs to terminate SRV cycling IAW EOP ATWS.

d. With the SRVs opening and closing, RPV level control becomes very difficult. Lower RPV level to less than +15 inches IAW EOP ATWS to slow the opening and closing of the SRVs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	239002	2.1.23
	Importance Rating		4.4

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (SRVs)

Proposed Question: SRO Question #89

Proposed Answer: C

A: Incorrect – Systems are operating as designed however the EOP at step P-3 states “Manually open SRVs to terminate SRV cycling”.

B: Incorrect – HPCI/RCIC can not go into CST-CST mode for pressure control with an initiation signal present (<119.5”)..

C: Correct – Per EOP ATWS Page 55 discussion of Step /P-3. Step directs “Manually open SRVs to terminate cycling”. Embedded in the bases is the definition of “Cycling”: multiple sequenced valve actuations with valve opening being initiated in response to RPV pressure increasing to or above the lifting setpoint and valve closure being governed by RPV pressure decreasing to or below the reset setpoint. The concerns with cycling are also stated including exerting significant dynamic loads on the SRV tailpipes and support structures.

D: Incorrect – lowering level is a concern, however, lowering level is not required action to stop the SRV’s from cycling.

Technical Reference(s): EOP ATWS Bases Rev 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: **DO NOT PROVIDE EOP ATWS /P LEG**

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

15. The plant is operating at full power. All systems are operable.

    You are provided with the following information:

    - SBLC Tank Concentration is 13.4%
    - SBLC Volume 2500 gallons
    - SBLC pump suction piping and tank temperature is 64°F
    - Attached TS LCO and associated figures

    Based on the above the SBLC system is __(1)__ because __(2)__.

    a. (1) operable
       (2) parameters are within required limits

    b. (1) inoperable
       (2) there is potentially an inadequate concentration of boron ONLY

    c. (1) inoperable
       (2) there is inadequate volume of boron solution ONLY

    d. (1) inoperable
       (2) there is an inadequate volume of boron and an inadequate boron concentration

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	211000	2.2.25
	Importance Rating		4.2

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (SLC)

Proposed Question: SRO Question #90

Proposed Answer: B

A: Incorrect – IAW TS Figure 3.1.7.1-2, the temperature is too low for the concentration, boron could potentially precipitate from the solution.

B: Correct - IAW TS Figure 3.1.7.1-2, the temperature is too low for the concentration, therefore, boron could potentially precipitate from the solution. If boron were to precipitate, the concentration would lower, resulting in a potentially inadequate concentration of boron per TS Figure 3.1.7-1.

C: Incorrect - IAW TS Figure 3.1.7.1-1, the volume of 2500 gal is adequate.

D: Incorrect - IAW TS Figure 3.1.7.1-1, the volume of 2500 gal is adequate.

Technical Reference(s): TS bases 3.1.7
TS 3.1.7 & figures (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.1.7 w/ figures

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2, 6

(2) Facility operating limitations in the technical specifications and their bases.
(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

16. The plant is operating at 90% power.

The following rods have been declared slow based on scram time testing: 14-23, 14-27 and 18-39.

At 1430 today, control rod 18-23 receives an accumulator alarm 1C05A (F-7), CRD ACCUMULATOR LO OR HI LEVEL.

An operator sent to investigate reports that, when the local panel pushbutton was depressed for HCU 18-23, the local alarm light remains lit for that HCU.

Based on the operator report, what caused the accumulator alarm and what, if any, action(s) would be in compliance with Technical Specifications?

a. The accumulator has a high water level.
Declare the control rod slow within 8 hours, be in MODE 3 within the following 12 hours.

b. The accumulator has a high water level.
Declare the control rod inoperable within 8 hours. Once declared inoperable, the control rod is required to be fully inserted AND disarmed within 3 hours.

c. The accumulator has a low pressure.
If accumulator pressure is <940 psig, declare the control rod slow within 8 hours, be in MODE 3 within the following 12 hours.

d. The accumulator has a low pressure.
If accumulator pressure is <940 psig, declare the control rod inoperable within 8 hours. Once declared inoperable, the control rod is required to be fully inserted AND disarmed within 3 hours.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201003	A2.08
	Importance Rating		3.7

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: Low HCU accumulator pressure/high level

Proposed Question: SRO Question #91

Proposed Answer: C

A: Incorrect – the cause of the alarm is low pressure

B: Incorrect – Per TS 3.1.3 – If declared inoperable, the rod must be fully inserted within 3 hours and disarmed within 4 hours.

C: Correct – Per SD 255 page 26 - The alarms for low nitrogen pressure and accumulator leakage are also annunciated on the local accumulator alarm panels 1C054 and 1C072. The alarm panels consist of a pushbutton for each accumulator that lights up when either low nitrogen pressure or accumulator piston leakage is detected. *If the light stays energized when the pushbutton is depressed, the originating signal is low nitrogen pressure*; if the light de-energizes when the pushbutton is depressed, the accumulator water level switch is actuated.

Per TS 3.1.5 - With One control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 900 psig, Declare the associated control rod scram time "slow." OR Declare the associated control rod inoperable.

Per TS 3.1.4 - No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations. If this rod were declared slow, 3 OPERABLE control rods that are "slow" would occupy adjacent locations. Therefore, the LCO applies to be in Mode 3 within 12 hours

D: Incorrect - Per TS 3.1.3 – If declared inoperable, the rod must be fully inserted within 3 hours and disarmed within 4 hours.

Technical Reference(s): TS 3.1.3, 3.1.4, 3.1.5
System Description 255, pg 26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.1.3, 3.1.4, 3.1.5
Core map

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

17. A normal plant Shutdown and Cooldown from full power operations is in progress.

- The reactor is shutdown
- RPV Pressure is 940 psig
- RPV level is 190 inches and stable
- Cooldown has just started
- The GEMAC Reference Leg Backfill System has been out of service for 3 weeks

(1) Which of the following is the correct directions for these plant conditions

AND

(2) Which procedure gives direction if “NOTCHING” is observed?

a. (1) When RPV pressure reaches 500 psig direct the operating crew **NOT** to use the Yarway instruments on 1C05 for level indication.

(2) OI 880 “Non-Nuclear Instrumentation System.”

b. (1) When RPV pressure reaches 500 psig direct the operating crew **NOT** to use the Yarway Instruments on 1C05 for level indication.

(2) IPOI 4 “Shutdown.”

c. (1) Direct enhanced RPV Level monitoring during the Shutdown.

(2) OI 880 “Non-Nuclear Instrumentation System.”

d. (1) Direct enhanced RPV Level monitoring during the Shutdown.

(2) IPOI 4 “Shutdown.”

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	216000	A2.11
	Importance Rating		3.3

Nuclear Boiler Instrumentation - Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION system and (b) based on those predictions, use procedures to correct, control , or mitigate the consequences of those abnormal conditions or operations: Heatup or cooldown of the reactor vessel

Proposed Question: SRO Question #92

Proposed Answer: C

A: Yarways share instrument lines with the GEMAC instruments but these directions would be given for a rapid pressure decrease. OI 880 is the correct OI for the action required.

B: Yarways share instrument lines with the GEMAC instruments but these directions would be given for a rapid pressure decrease. IPOI 4 does direct action be performed IAW OI 880 for these conditions but does not have directions if Notching is observed.

C: Anytime the Reference Leg Backfill system is out of service for 7 days performance of OI-880, J-1 section 6.1 is required. Notching is expected when reducing RPV pressure during a shutdown. The Narrow range GEMAC level instruments are susceptible. OI-880 directs these actions for the given plant conditions.

D: IPOI 4 does direct action be performed IAW OI 880 for these conditions but does not have directions if Notching is observed. Enhanced level monitoring is the correct direction for these conditions.

Technical Reference(s): OI-880 Rev 11 pages 3 & 14; IPOI 4 Rev 68 P&L 25 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SRO 4.18.03, Direct operator actions to control RPV level throughout the cooldown. (As available)

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

Question History: Last NRC Exam: Yes, 2002 NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 5

(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

18. The plant is operating at full power. A radiological event on the refuel floor causes a release.

    Then, annunciator 1C-07A (D-11), Control Building HVAC Panel 1C-26 Trouble, alarms.

    Operators are dispatched to investigate the alarm. They report the following two 1C-26 alarms:

    - 1C26A (C-2), Control BLDG Intake Air Rad Mon RIM-6101A Hi/Trouble
    - 1C26B (C-2), Control BLDG Intake Air Rad Mon RIM-6101B Hi/Trouble

    Which one of the following describes the effects on Control Building ventilation and action that is required?

    a. A Control Building isolation should have occurred. Verify only one Battery Exhaust fan is running IAW OI 730, Control Building HVAC System.

    b. A Control Building isolation should have occurred. Verify two Battery Exhaust fans are running IAW OI 730, Control Building HVAC System.

    c. Verify that Control Building pressure is being maintained at a negative value. Verify only one Battery Exhaust fan is running IAW ARP 1C26A & B (C-2).

    d. Verify that Control Building pressure is being maintained at a positive value. Verify two Battery Exhaust fans are running IAW ARP 1C26A & B (C-2).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	272000	2.1.31
	Importance Rating		4.3

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (Radiation Monitoring)

Proposed Question: SRO Question # 93

Proposed Answer: A

Explanation (Optional): KA Justification – This KA is typically used for scenario/JPM evaluation. In this case a question was asked which requires the ability to determine control room indication given an event and then determine how the indications reflect the control room ventilation lineup and pressure. Additionally, the applicant must determine the appropriate action to be taken for the event.

A. Correct – Per OI 730 P&L 9, page 5, to maintain positive pressure during a control building isolation, only ONE battery exhaust fan shall be running.
ARP 1C26A & B (C-2) contains the same information.

B. Incorrect – Only one fan shall be running.

C. Incorrect – Positive pressure shall be maintained.

D. Incorrect - Positive pressure shall be maintained. Only one fan shall be running

Technical Reference(s): OI 730 Rev 100 P&L #9 page 5
ARP 1C26A & B (C-2) Rev 48 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 4, 5

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

19. While supervising fuel handling activities in the Spent Fuel Pool, you discover a minor typographical error in the approved Fuel Moving Plan (FMP) that you are using.

    The final orientation for the spent fuel bundle being moved is illegible.

    Which of the following describes the process for correcting the error to the fuel moving plan?

    a. Minor pen & ink changes to the FMP may be made by the Fuel Handling Supervisor with concurrence from the Shift Manager.

    b. Any changes in the FMP require a Procedure Change Request initiated by Reactor Engineering with concurrence from the Fuel Handling Supervisor and the Shift Manager.

    c. Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

    d. Minor pen & ink changes to the FMP may be made by the Fuel Handling Supervisor with concurrence from Reactor Engineering. The Shift Manager must be advised but Shift Manager concurrence is NOT required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.1.40
	Importance Rating		3.9

Knowledge of refueling administrative requirements

Proposed Question: SRO Question # 94

Proposed Answer: C

A: Incorrect – Concurrence is required by Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

B: Incorrect – A procedure change request is not required.

C: Correct – PER RFP 4-3. Step 5.1.1.e - Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

D: Incorrect - Concurrence is required by Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

Technical Reference(s): RFP 403 Rev 33 Step 5.1.1.e. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: Fuel handling 1.4.1.1. (As available)

Question Source: Bank # DAEC 22624
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 7

(7) Fuel handling facilities and procedures.

1 Point

20. System engineering has proposed a new performance test on the RCIC pump which will affect pump flow rate. Engineering has determined that the Technical Specification for pump flow would not be adversely affected during the test.

    IAW ACP 1407.4, Special Test Procedures (SpTP), which one of the following describes how the test is classified and who must provide written authorization for the SpTP.

    a. This test is considered an Infrequently Performed Test or Evolution AND a Special Test.
       The Plant Manager and the CRS.

    b. This test is considered ONLY a Special Test.
       The Plant Manager and the CRS.

    c. This test is considered an Infrequently Performed Test or Evolution AND a Special Test.
       ONLY the on-shift CRS.

    d. This test is considered ONLY a Special Test.
       ONLY the on-shift CRS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #		2.2.7
	Importance Rating		3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question: SRO Question # 95

Proposed Answer: C

A: Incorrect – Any Special Test is also considered an Infrequently Performed Test or Evolution. Although the test may be reviewed by the Plant Manager, their written approval is not required prior to on shift performance

B: Incorrect - Any Special Test is also considered an Infrequently Performed Test or Evolution AND a Special Test

C: Correct – Per ACP 1407.4 - Special Test or Experiment - Non-routine operations performed to determine the performance characteristics of a structure, system or component. Special Tests are non-routine tests that are not required by the Technical Specifications, a 10CFR 72 Certificate of Compliance, or the ASME Section XI Manual, and are not described in the UFSAR or a 10CFR 72 Final Safety Analysis Report (Certificate Holder's), as updated.

Per ACP 1407.4 Step 3.3 (10) - SpTPs are considered Infrequently Performed Test or Evolutions (IPTEs). Refer to ACP 102.17, Pre/Post-Job Briefs and Infrequently Performed Tests and Evolutions, for IPTE requirements.

Per ACP 1407.4 Step 3.5 (3) - All SpTPs require written authorization from the on-shift CRS prior to performance.

D: Incorrect - Any Special Test is also considered an Infrequently Performed Test or Evolution AND a Special Test

Technical Reference(s): ACP 1407.4 Rev 21 Definitions, Steps 3.5 (3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content:		
	55.41	
	55.43	1

(1) Conditions and limitations in the facility license.

1 Point

21. With the plant in MODE 1, an Outboard Primary Containment Isolation Valve, required to be operable in MODES 1, 2 and 3, failed its stroke time testing. To comply with the associated LCO, the inoperable valve has been CLOSED and DEACTIVATED.

    Which ONE of the following describes the conditions REQUIRED for Post Maintenance Testing to restore OPERABILITY, which includes electrically stroking this valve?

    a. This valve CANNOT be electrically stroked until the plant is in MODE 4, COLD SHUTDOWN, when the valve is not required to be operable.

    b. This valve may be electrically stroked under Administrative Control without regard to the position of the other isolation valve in the same line.

    c. This valve may ONLY be electrically stroked if the INBOARD valve in the same line is CLOSED.

    d. This valve may ONLY be electrically stroked if the valve is reclosed within 4 hours IAW Technical Specifications.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #		2.2.21
	Importance Rating		4.1

Knowledge of pre- and post-maintenance operability requirements

Proposed Question: SRO Question # 96

Proposed Answer: B

A: Incorrect – In MODE 4, Primary Containment Isolation Valve OPERABILITY is NOT APPLICABLE. It is not required to shutdown to stroke this valve.

B: Correct – Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

C: Incorrect – Redundant valve closure is an acceptable method to allow valve stroking, but it is not the ONLY acceptable method.

D: Incorrect – There is no requirement to have the valve reclosed within 4 hours of opening it. The requirement is to have administrative control of the valve opening.

Technical Reference(s): TS LCO 3.0.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # WTS – 2496
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

1 Point

22. The plant is in MODE 5, with the following:

- Fuel Movements are in progress between the cavity and the fuel pool
- SDC Cooling Isolation Valve MO-1909 spuriously closed and is jammed on its closed seat
- Shutdown Cooling Flow has been secured for 2 hours
- Maintenance is working on several of the outboard MSIVs
- Reactor Coolant temperature is 105 degrees F.

Which one of the following actions will result in meeting Technical Specification requirements for an alternate means of decay heat removal?

a. Start a Recirc Pump immediately regardless of the core configuration IAW OI 264, Reactor Recirculation System, to provide forced circulation.

b. Raise reactor water level and control it between 230 and 240 inches as measured on the GEMACs IAW AOP 149, Loss of Decay Heat Removal. Increase CRD flow to enhance natural circulation.

c. Establish Feed and Bleed to the Torus via the SRVs IAW OI 183.1, Automatic Depressurization System. Ensure all personnel are cleared from the Torus.

d. Align Fuel Pool Cooling return to the vessel cavity IAW AOP 149, Loss of Decay Heat Removal. RBCCW flow and cooling must be maximized.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #		2.4.9
	Importance Rating		4.2

Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies

Proposed Question: SRO Question # 97

Proposed Answer: D

A: Incorrect – Per AOP 149 this is not defined as an alternate means of decay heat removal to satisfy TS.

B: Incorrect – Cavity is already flooded to the weirs and Floodup level indication is used, not GEMACS

C: Incorrect – Not an acceptable method because steam line plugs are installed

D: Correct – This is a prescribed method in AOP 149 Section 4.5

Technical Reference(s): AOP 149 Rev 31
TS 3.9.7.Bases A.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 2,5

(2) Facility operating limitations in the technical specifications and their bases.
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

23. The plant was initially operating at full power. A fuel leak resulted in high Offgas and Main Steam Line Radiation Levels.

    AOP 672.2, Offgas Radiation, Reactor Coolant High Activity has been entered and a plant shutdown is being performed to comply with Technical Specifications.

    Then, a spurious Main Turbine trip occurred and the plant automatically scrammed.

    Plant conditions are as follows:

    - ALL Control Rods are fully inserted
    - Reactor level lowered to 160” following the scram and is now stable at 184”
    - Reactor Pressure is 920 psig with the Turbine Bypass Valves in service
    - Offgas is in service, maintaining 2 inches Hg Backpressure
    - 1C05B C-2 MAIN STEAM LINE HI HI RAD / INOP TRIP continues to alarm

    With these conditions, which one of the following actions are required and will MINIMIZE release of radioactivity to the environment?

    a. Enter EOP 1, RPV Control, and maintain RPV level 170” to 211”. No additional EOP entries are required at this time.
    Cooldown at LESS THAN 100°F/hr by depressurizing to the Main Condenser to allow the Offgas treatment process to limit radioactivity releases.

    b. Enter EOP 1, RPV Control, and EOP 4, Radioactivity Release Control.
    Rapidly cooldown at GREATER THAN 100°F/hr by depressurizing to the Main Condenser to allow the Offgas treatment process to limit radioactivity releases.

    c. Enter EOP 1, RPV Control, and maintain RPV level 170” to 211”. No additional EOP entries are required at this time.
    Cooldown at LESS THAN 100°F/hr by depressurizing to the Torus to allow the Containment to limit radioactivity release and allow the Main Condenser to be used to control MSIV Leakage.

    d. Enter EOP 1, RPV Control, and EOP 4, Radioactivity Release Control.
    Rapidly cooldown at GREATER THAN 100°F/hr by depressurizing to the Torus to allow the Containment to limit radioactivity release and allow the Main Condenser to be used to control MSIV Leakage.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #		2.3.11
	Importance Rating		4.3

Ability to control radiation releases

Proposed Question: SRO Question # 98

Proposed Answer: C

A: Incorrect – Action would be correct for a normal shutdown without High RCS Activity concerns.

B: Incorrect – Action would be correct if Emergency Depressurization were anticipated during EOP execution. No reasons are provided in stem for ED

C: Correct - AOP 672.2, Off Gas Radiation, Reactor Coolant High Activity specifies closing the MSIVs and MSL Drains, depressurizing to the Torus. Main Steam and Main Condenser will be aligned to limit MSIV Leakage. NO requirement has been given to Anticipate Emergency Depressurization, so normal cooldown limits are in effect.
EOP -1 entry required on low RPV level, IPOI 5 entry not required because the scram already occurred (EOP 1 Decision Step RC-2)
No other EOP entries exist.

D: Incorrect – Action would be correct if Emergency Depressurization were required and if EOP-4 Radioactivity Release Control, were entered. No entry conditions for these are given in stem

Technical Reference(s): AOP 672.2 Rev 33 Step 6
EOP – 1
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # WTS – 2499
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 4, 5

(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1 Point

24. An event has occurred at the plant. The TSC and EOF are activated but NOT yet operational.

    IAW Emergency Plan Implementing Procedures, which one of the following is the individual responsible for escalating an emergency event level from a Site Area Emergency to a General Emergency?

    a. Shift Manager

    b. Operations Manager

    c. Plant Manager

    d. Site Vice President

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #		2.4.38
	Importance Rating		4.4

Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Proposed Question: SRO Question # 99

Proposed Answer: A

A: Correct- Per EPIP 2.5 – Step 3.1 (1) - Upon determining that the plant is in an unexpected operational condition, the Operations Shift Manager/Control Room Supervisor (OSM/CRS) shall evaluate plant conditions using guidance contained in EPIP 1.1, "Determination of the Emergency Action Level," and, as warranted, classify the event in one of the four emergency categories.

Per Step 3.1.(2).(a) - The OSM/CRS shall function additionally as the Emergency Coordinator and Site Radiation Protection Coordinator until relieved of such function by appropriately qualified personnel.

Until the TSC and EOF are operational, the SM retains the responsibility of escalating the event.

B: Incorrect – The SM/CRS is the EC until the other facilities are operational.

C: Incorrect – The Emergency Response & Recovery Director would be responsible if the EOF were operational

D: Incorrect – The Site VP is not designated as the EC for the described situation.

Technical Reference(s): EPIP 2.5 Rev 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam: No

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

10 CFR Part 55 Content: 55.41
55.43 1
(2) Facility operating limitations in the technical specifications and their bases.

1 Point

25. It is 0400 and the plant is in Hot Shutdown. The STA is informed by their spouse that they must return home immediately for a family emergency.

- At 0405, the STA departs as directed by the Shift Manager (SM).
- At 0410, the SM calls the Operations Manager to inform him of the reduction in crew composition.
- At 0420, the SM reaches a relief for the STA and directs him to come to work.
- At 0615, the STA relief arrives and joins the SM/CRS turnover.
- At 0645, the STA shift turnover briefing is completed.

Which one of the following describes the SM compliance with the shift manning requirements IAW ACP 1410.1, Conduct of Operations and Technical Specifications?

a. The shift manning requirements have been fully complied with because the STA function is ONLY required during Power Operation and Startup.

b. The shift manning requirements have NOT been fully complied with because the STA function was vacant for more than 2 hours.

c. The shift manning requirements have been fully complied with because the relief STA received a complete turnover within 4 hours of the previous STA departure.

d. The shift manning requirements have NOT been fully complied with because the Plant Manager's permission must be obtained before shift staffing drops below minimum requirements.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.1.5
	Importance Rating		3.9

Ability to use procedures related to shift staffing, minimum crew complement, overtime limitation, etc.

Proposed Question: SRO Question #100

Proposed Answer: B

A: Incorrect – Per TS 5.2.2.c – ONLY 2 hours is permitted for a shift staffing vacancy. Per ACP 1410.1 and TS the STA is required during Modes 1,2 and 3

B: Correct – Per TS 5.2.2.c - Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

Per ACP 1410.1 Section 3.2(3) - When the reactor is in other than COLD SHUTDOWN or REFUEL, the operations supervision team shall consist of at least three individuals. At any one time, there shall be at least one individual qualified to perform the OSM duties, at least one individual qualified to perform the CRS duties, and at least one individual qualified to perform the STA function on the operating crew.

C: Incorrect – The time limitation is 2 hours not 4 hours

D: Incorrect – The Operations Manager permission is required not the Plant Manager

Technical Reference(s): ACP 1410.1 rev 71
TS 5.2.2.c
TS 5.2.2.g
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: NONE

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2001

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

10 CFR Part 55 Content: 55.41
55.43 2

(2) Facility operating limitations in the technical specifications and their bases.

The following references were provided to the SRO applicants for use during the exam:

- Torus Level, Torus Temperature, and Primary Containment Pressure control legs of EOP-2, Primary Containment Control flowchart.
- Level control leg of the ATWS RPV Control flowchart with the set-points redacted.
- Pressure Suppression Pressure and Heat Capacity Limit curves from EOP-2
- EAL Matrix
- DAEC Core Map Showing Core Component Location
- Technical Specification LCO's
    - 3.1.3
    - 3.1.4
    - 3.1.5
    - 3.1.7 with associated figures