Examination Number/Title: PDA 17-1 NRC Exam RO, Rev. 0		
Training Program: Operations		
Course/Lesson Plan Number(s): 50007		
Total Points Possible: 75	**PASS CRITERIA: ≥ 80%**	**Exam Time: 360**

	Yes	No		Yes	No
This is an alternate examination; verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified > 50% difference.	☐	☒	This is a remediation exam. Verified the questions are different from the failed exam by at least the following criteria listed below: • 70% for Maintenance/Technical • 90% for Operations training programs	☐	☒
This is an initial training examination; verified at least 30% of the questions are different from previous administration of the same exam.	☒	☐	This is a LOCT annual operating exam or biennial comprehensive remedial exam, verified the questions are 100% different from the failed exam.	☐	☒
This is a non-randomly generated exam from an electronic exam bank, printed out or administered online. Verified at least 30% of the questions are different from other forms/versions of this exam (e.g., Forms A, B, C; continuing training exam versions for consecutive weeks). For LOCT weekly exams during a segment, verified > 50% difference.	☐	☒	This is a randomly generated exam from an electronic exam bank, printed out or administered online. Verified the exam bank has 3 questions per objective if one test item on exam for the objective. If 2 or more test items on exam for an objective, then 6 questions are in bank.	☐	☒
NOTE: • See TR-AA-230-1003, SAT Development, for exam development and review guidelines. • NRC exams may require additional information. Refer to fleet and site specific procedures.			Key should contain the following: ▪ Learning Objective Number ▪ Test Item ○ Question or Statement ○ All possible answers ○ Correct Answer Indicated ○ Point Value ○ References (if applicable)		

EXAMINATION REVIEW AND APPROVAL:	
Developed by:	**Date:**
Instructional Review of Written Exam (Qualified Instructor):	**Date:**
Technical Review (SME):	**Date:**
Approved by Training Supervisor:	**Date:**
Approved by Training Program Owner (or line designee):	**Date:**

Indicate in the following table if any changes are made to the exam after approval:

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR# (if applicable)	PREPARER / SUPERVISOR	DATE / DATE

NEXTera ENERGY NUCLEAR FLEET	WRITTEN / ORAL EXAMINATION COVER SHEET	Page 1

Trainee Name:			
Employee Number:		**Site:**	**PDA**
Examination Number/Title: PDA 17-1 NRC Exam RO, Rev. 0			
Training Program: Operations			
Course/Lesson Plan Number(s): 50007			
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 minutes** to complete the examination.
7.	Feedback on this exam may be documented on TR-AA-230-1004-F03, Examination 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:**

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK3.03	
	Importance Rating	2.8	

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Idle Loop Flow

Proposed Question: RO Question 1

- The plant is operating in single loop
- The running Recirc Pump speed is 55%

Idle Loop Flow is in the ___(1)___ flow direction and this flow is ___(2)___ Total Core Flow?

	(1)	(2)
A.	forward	added to
B.	forward	subtracted from
C.	reverse	added to
D.	reverse	subtracted from

Proposed Answer: D

Explanation (Optional):

A. Incorrect: This would be correct if both recirc pumps were operating.

B. Incorrect: This would be correct if the running recirc pump was operating at <50% pump speed. With the MG Set field breaker tripped open (one recirc pump is not running,) the flow is being correctly subtracted from Total Core Flow.

C. Incorrect: This would be correct if the MG Set field breaker was closed or the idle loop discharge valve was closed. To keep the idle loop warm the Operating Instruction will direct that the idle loop discharge and discharge bypass valves to be open.

D. Correct: At this pump speed there is reverse flow in the idle jet pumps. OI 264, Recirculation Pump System, NOTE for single loop operations has idle loop flow transitions from forward to reverse flow at approximately 50% drive flow. SD-264 discusses that the FR-4502 is interlocked with the recirculation pump generator field breaker contacts and recirculation pump discharge valve position indications.

Technical Reference(s): OI -264, Rev. 138 (Attach if not previously provided)

SD-264, Rev. 13

Proposed References to be provided to applicants during examination: N

Learning Objective:	12.00.00.07, explain the derivation on Total Recirc Flow and Total Core Flow, include in the explanation: d. subtracting circuitry	(As available)

Question Source: Bank #

Modified Bank # X (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

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

Comments: Modified 2013 NRC Exam question. Raised recirculation speed > 50% this makes answer choice D correct and B incorrect.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.01	
	Importance Rating	3.4	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Cause of partial or complete loss of A.C. power

Proposed Question: RO Question 2

- The plant is operating at full reactor power
- 1A3, 4160 VAC Essential Switchgear, bus is de-energized

NOTE: On page 4 is a picture of annunciators from 1C08.

The Operator will restore power using the _______.

A. Auxiliary Transformer

B. "B" Diesel Generator

C. Startup Transformer

D. Standby Transformer

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The Auxiliary Transformer, which is available as indicated annunciators 1C08A(A-2) and 1C08B(A-9), however this transformer is not electrically designed to align to support electrical power to the 1A3 and 1A4 essential electrical buses.

B. Incorrect: The "B" Diesel Generator, which is available, is not a swing diesel generator which can be aligned to supply electrical power to the 1A3 essential electrical bus.

C. Incorrect: Annunciator 1C08(C-7) demonstrates that the Startup Transformer has a lockout and is not available to supply electrical power to either 1A3 or 1A4.

D. Correct: Annunciator 1C08A(C-6) is not received and indicates that electrical power can be supplied to 1A3 Essential Electrical bus.

Technical Reference(s): SD-304, Rev. 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	15.00.00.02, Describe all allowed electrical lineups for the Essential Electrical Distribution System	(As available)

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

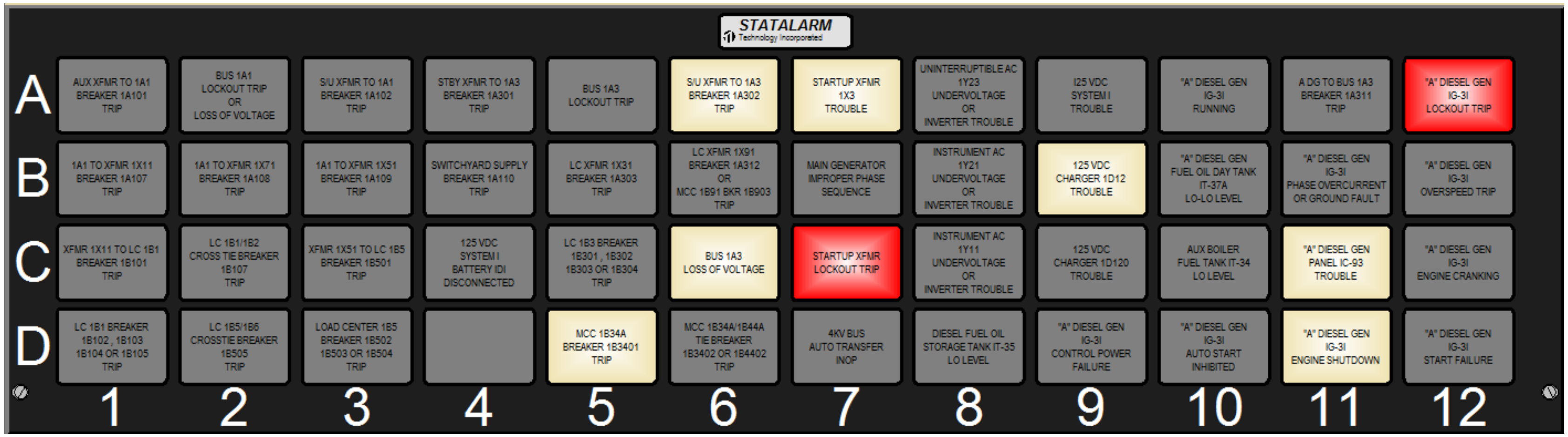
Question History: Last NRC Exam:

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

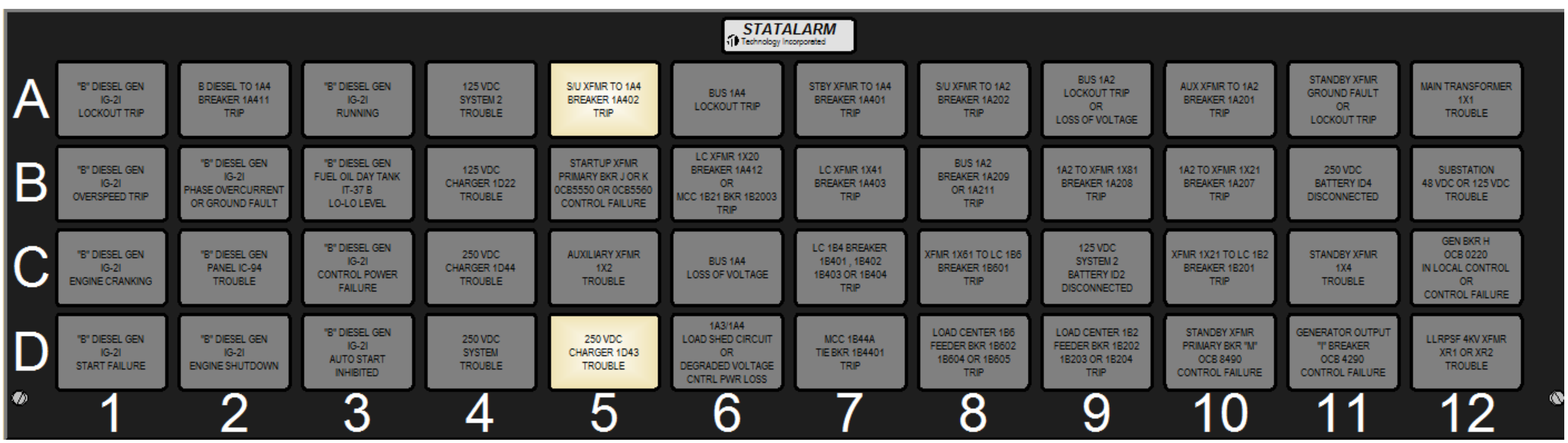
10 CFR Part 55 Content:	55.41	10
	55.43	

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

This picture is for RO Question 2



1C08A



1C08B

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK3.03	
	Importance Rating	3.1	

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Reactor SCRAM: Plant-Specific

Proposed Question: RO Question 3

- The plant was operating at full reactor power
- A loss of 125 VDC DIST PANEL 1D10 occurs
- No Operator action is taken

An automatic reactor scram will occur because of?

A. A turbine trip

B. MSIV Closure

C. Reactor Recirc Pumps Trip

D. Loss of 'B' Feedwater Inverter

Proposed Answer: A

Explanation (Optional):

A. Correct – A failure downscale of both Rx 'B' Feedwater flow and steam flow will cause Feedwater control to open the Feed Reg Valves causing Rx water level to go high and thus causing a turbine trip and therefore, a Rx Scram.

B. Incorrect – A loss of the 'B' Feedwater inverter will only occur if a loss of Div II 125VDC power occurs.

C. Incorrect – an MSIV closure will not occur with a loss of DC only. MSIV closure in terms of loss of power would need a loss of both AC and DC power to occur.

D. Incorrect – Reactor Recirc Pumps trip is a result of the reactor water level rising to trip the Main Turbine (211 inches.)

Technical Reference(s): AOP 302.1 Rev. 57 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.06.01.07, Relate how each step and its performance meets the mitigation strategies of AOP 302.1	(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	

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 G2.4.11	
	Importance Rating	4.0	

Knowledge of abnormal condition procedures: Main Turbine Trip

Proposed Question: RO Question 4

- The plant is operating at full reactor power
- A large lube oil leak has developed near the Main Generator
- 1C07A (A-7), Turbine Lube Oil Bearing Header LO Pressure, alarms
- The in plant Operator CANNOT maintain lube Oil Tank level

Which actions are required by AOP 693, Main Turbine/EHC Failures?

The ___(1)___ and the condenser vacuum shall be ____(2)____.

A. (1) Main Turbine is tripped, the automatic reactor scram verified
(2) broken

B. (1) Main Turbine is tripped, the automatic reactor scram verified
(2) maintained

C. (1) reactor will be scrammed then the Main Turbine manually tripped
(2) broken

D. (1) reactor will be scrammed then the Main Turbine manually tripped
(2) maintained

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The reactor is scrammed first prior to tripping the Main Turbine.

B. Incorrect: The reactor is scrammed first prior to tripping the Main Turbine. Main condenser vacuum must be broken to allow the Main Turbine to stop in the shortest amount of time.

C. Correct: The reactor is scrammed, then the turbine is tripped, MSIV's are closed and the Main Condenser vacuum is broken.

D. Incorrect: Main condenser vacuum must be broken to allow the Main Turbine to stop in the shortest amount of time.

Technical Reference(s): AOP 693, Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:			N
Learning Objective:	94.49.00.03, Evaluate plant conditions and control room indications to determine the required operator actions		(As available)
Question Source:	Bank #	50189	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	PDA 2011
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AA1.01	
	Importance Rating	4.2	

Ability to operate and/or monitor the following as they apply to SCRAM: RPS

Proposed Question: RO Question 5

Which of the following would require an Operator to take manual action to insert all control rods?

A. Turbine trip at 40% reactor power

B. Reactor power spikes to 16% in MODE 2

C. Torus water level lowered to 6 feet in MODE 2

D. Steam tunnel temperature rises to 2100F in MODE 1

Proposed Answer: C

Explanation (Optional):

A. Incorrect: With the reactor producing steam at greater than 26% (as measured from main turbine first stage pressure) the RPS scram logic is no longer bypassed and an automatic reactor scram will be inserted.

B. Incorrect: With the MODE switch in MODE 2, the APRM scram signal is enforced at 15% reactor power. This will insert an automatic reactor scram.

C. Correct: With torus water level below 7.1 feet and in MODE 2, EOP 2 will require the Operating crew to insert a manual reactor scram. There are no automatic reactor scram signals on Torus water level.

D. Incorrect: With steam tunnel ambient temperature greater than 2000F the Main Steam Isolation valves will receive a closed signal. With the MSIV's at less than 90% open will insert an automatic scram signal.

Technical Reference(s): Bases-EOP 2, Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 22.00.00.03, list the signals which cause a Reactor Protection System trip including setpoints and logic, and (As available)

describe how they are bypassed and
how they are reset

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AA2.03	
	Importance Rating	4.3	

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor pressure

Proposed Question: RO Question 6

- The plant was operating at full reactor power
- A NON-FIRE event occurred which required Control Room abandonment
- A manual scram was inserted
- All control rods were verified FULL IN
- The Reactor Operator has completed the “as time permits” actions of AOP 915
- The Control Room has been evacuated

What position is the MODE switch in when evacuation was completed?

A. RUN

B. REFUEL

C. SHUTDOWN

D. START & HOT STBY

Proposed Answer: A

Explanation (Optional):

A. Correct: AOP 915 requires the MODE Switch to be placed in RUN if it was in RUN originally. This is to ensure that the 850 psig in “RUN” Main Steam Isolation Valve closure logic can respond to stop a lowering reactor vessel pressure trend through main steam.

B. Incorrect: The MODE Switch would have been placed in REFUEL to verify the control rods were full in and then placed in SHUTDOWN for a normal manual reactor scram.

C. Incorrect: The MODE Switch would have been placed in SHUTDOWN if a normal reactor scram was inserted. With the control room abandonment, the MODE Switch would have been returned to RUN.

D. Incorrect: With the control room abandonment, the MODE Switch would have been returned to RUN.

Technical Reference(s): AOP 915, Rev. 57 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.28.01.07, Explain how each step and its performance meets the mitigation strategies of AOP 915	(As available)

Question Source:	Bank #	20296	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	PDA 13-1 EOP Comprehensive	Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AK1.01	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to [with a] PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Proposed Question: RO Question 7

- The plant is operating at full reactor power
- The "A" SBDG is inoperable
- The engine fuel racks are tripped

An event occurred and a manual reactor scram was inserted.

- Drywell pressure is 3.2 psig and rising
- Reactor water level is 145 inches and rising

"A" side ECCS pumps ______.

A. did **NOT** start and cannot be started due to a loss of electrical power

B. have started and may be operated with no additional Operator action

C. have started but "A" ESW pump must be manually started for continued operation

D. did **NOT** start but may be operated normally after the "A" ESW pump is manually started

Proposed Answer: C

Explanation (Optional):

A. Incorrect: "A" ECCS pumps will start on a 2 psig drywell pressure signal.

B. Incorrect: "A" ESW flow is required for continued operation for the "A" ECCS pumps.

C. Correct: With the fuel rack tripped, the "A" ESW pump will not auto start but can be manually started. "A" ECCS pumps will start on a 2 psig drywell pressure signal. "A" ESW flow is required for continued operation for the "A" ECCS pumps.

D. Incorrect: A" ECCS pumps will start on a 2 psig drywell pressure signal.

Technical Reference(s): OI 324, Rev. 119 (Attach if not previously provided)

ARP 1C05B(A-1), Rev. 106

Proposed References to be provided to applicants during examination: N

Learning Objective:	33.00.00.02, List the signals which cause an ESW System auto initiation including setpoints and logic. Describe how they are bypassed and how they are reset.	(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	8
	55.43	

Comments: Components, capacity, and functions of emergency systems.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 G2.1.30	
	Importance Rating	4.4	

Ability to locate and operate components, including local controls: Partial or Complete Loss of Instrument Air

Proposed Question: RO Question 8

- The plant is operating at full reactor power
- The crew has entered AOP 518, Failure of Instrument and Service Air

The Operator will have to go to the _____(1)_____ to place 1K001, Backup Instrument Air Compressor, in PRIMARY.

A. Turbine Building Basement

B. Reactor Building 2nd floor

C. Air Compressor Building

D. Intake Structure

Proposed Answer: A

Explanation (Optional):

A. Correct: The Operator will have to go to the Turbine Building basement to place 1K001, Backup Instrument Air Compressor, to primary.

B. Incorrect: This would be the correct location to operate the 1K003 and 1K004 air compressors.

C. Incorrect: This would be the correct location to operate the 1K090A/B/C air compressors.

D. Incorrect: This would be the correct location to operate the 1K016A/B air compressors.

Technical Reference(s): SD-518, Rev. 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 36.00.00.02, describe the major flow paths for the Instruments and Service (As available)

Air System, including, a. Major components

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AK1.01	
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Decay heat

Proposed Question: RO Question 9

- The reactor has been shutdown for 40 days
- RPV level is 200 inches and steady
- Shutdown cooling has just been lost
- Reactor coolant temperature is currently 120°F and rising
- Reactor recirculation pumps are not available

With the given conditions:

(1) What is the approximate time it will take to reach boiling in the reactor?

AND

(2) What alternate decay heat removal method is available given the current conditions?

A. (1) 6.1 hours
(2) Fuel Pool Cooling to the Reactor Vessel Cavity

B. (1) 6.1 hours
(2) Reactor Water Cleanup Heat Exchanger

C. (1) 31.7 hours
(2) Fuel Pool Cooling to the Reactor Vessel Cavity

D. (1) 31.7 hours
(2) Reactor Water Cleanup Heat Exchanger

Proposed Answer: B

Explanation (Optional):

A. Incorrect: if the correct graph is used (APPENDIX 2 - HEATUP RATE CURVE RPV LEVEL AT 200"), an approximate heatup rate of 40°F/hr will be obtained. (212°F - 120°F) / (15°F/hr) = 6.1°F/hr. Fuel Pool Cooling to the Reactor Vessel Cavity can only be accomplished if the reactor vessel is in a flooded condition.

B. Correct: if the correct graph is used (APPENDIX 2 - HEATUP RATE CURVE RPV LEVEL AT 200"), an approximate heatup rate of 40°F/hr will be obtained. (212°F - 120°F) / (15°F/hr) = 6.1°F/hr. The reactor water cleanup heat exchanger system has the capability to be used with the reactor water level at 200".

C. Incorrect: this answer would be reached if the incorrect graph was used (APPENDIX 1 - HEATUP RATE CURVE - RPV FLOODED). This would yield and approximate heatup rate of 2.9°F/hr. (212°F - 120°F) / (2.9°F/hr) = 31.7°F/hr. Fuel Pool Cooling to the Reactor Vessel Cavity can only be accomplished if the reactor vessel is in a flooded condition.

D. Incorrect: this answer would be reached if the incorrect graph was used (APPENDIX 1 - HEATUP RATE CURVE - RPV FLOODED). This would yield and approximate heatup rate of 2.9°F/hr. (212°F - 120°F) / (2.9°F/hr) = 31.7°F/hr. The reactor water cleanup heat exchanger system has the capability to be used with the reactor water level at 200".

Technical Reference(s): AOP 149 Rev.44 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: AOP-149, Appendix 1 (HEATUP RATE CURVE - RPV FLOODED) and Appendix 2 (HEATUP RATE CURVE - RPV LEVEL AT 200")

Learning Objective: 94.01.02.03 (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA2.04	
	Importance Rating	3.4	

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Occurrence of fuel handling accident

Proposed Question: RO Question 10

Which of the following is classified as a refueling accident?

A. An accidental criticality during shutdown margin testing

B. An accidental control rod withdrawal with the core fully loaded

C. An accidental dropping of a fuel assembly on the top of the core

D. An accidental control rod drop during the initial startup

Proposed Answer: C

Explanation (Optional):

A. Incorrect - As this would occur only with the primary containment intact and this would not provide as significant radiation release.

B. Incorrect - As this event is sufficiently restricted by refueling interlocks and administrative controls.

C. Correct - The DBA for a Fuel Handling Accident represents the event that releases the largest quantity of radioactive material directly to the Secondary Containment.

D. Incorrect - As this would occur only with the primary containment intact and this would not provide as significant radiation release.

Technical Reference(s): UFSAR 15.2.5 Rev. 23 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 94.55.01.01, Determine the actions required to mitigate a fuel handling event (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 10

55.43

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK2.03	
	Importance Rating	3.8	

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Low Pressure Core Spray System (LPCS): Plant-Specific

Proposed Question: RO Question 11

- The plant was operating at full reactor power
- The crew inserted a manual reactor scram
- Torus sprays are operating

Containment parameters are:

- Torus pressure is 5 psig and rising
- Torus water temperature is 900F and rising slowly
- Torus water level is 10.6 feet and rising

No additional Operator action is taken, Core Spray will ______________.

A. have improved Net Positive Suction Head as Drywell pressure continues to rise

B. have a lower discharge flow rate into containment as Drywell pressure continues to rise

C. be more susceptible to the system vortex limit as non-condensable gases build up in containment

D. be unaffected by these parameters since the suction path is normally aligned to the Condensate Storage Tank

Proposed Answer: A

Explanation (Optional):

A. Correct: The normal alignment of the Core Spray system is to the Torus. If Drywell pressure is allowed to continue to rise, the system NPSH will improve as the pump will be driven further away from cavitation. This concept is demonstrated with Bases-Curves, Figure 20.

B. Incorrect: If discharge pressure is allowed to rise for the Core Spray pump, the discharge flow rate will lower. However, the Drywell pressure rise is felt on the suction and the discharge of the Core Spray pump. The Drywell pressure rise will not effect system flow rate.

C. Incorrect: With a leak from the reactor vessel, more non-condensable are going to build up in containment. Air entrainment into the suction of the Core Spray pump is a concern

which the Vortex Limit is monitored by the crew to prevent. However, this limit is based on the Torus water level and Core Spray system flow rate.

D. Incorrect: The Core Spray system would be unaffected if the Core Spray suction was aligned to the Condensate Storage Tanks (CST.) The normal system alignment required by Technical Specifications requires the suction path to be aligned to the Torus. OI 151 does allow for alignment to the CST, but this is not a normal alignment.

Technical Reference(s): Bases-Curves, Rev. 16 (Attach if not previously provided)

SD-151, Rev. 7

Proposed References to be provided to applicants during examination: Core Spray Vortex Limit, Core Spray NPSH Limit

Learning Objective: 95.00.00.17, evaluate plant status and control room indications to determine the applicability and affects of any EOP curve or limit. (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK2.11	
	Importance Rating	3.5	

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:
Reactor water level

Proposed Question: RO Question 12

- The plant is operating at 40% reactor power
- “D” Inboard Main Steam Isolation Valve fails closed

During the pressure transient, reactor water level will initially ____________.

A. lower due to greater backpressure experienced on the feedwater regulating valves

B. lower due to the initial reduction of core voids and feedwater level control response

C. rise due to greater core plate differential pressure from the initial reactor power rise

D. rise due to feedwater level control sensing more steam flow in the “B” Main Steam line

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Reactor water level lowers due to a reduction of core void concentration in the fuel region.

B. Correct: When pressure rises within the reactor vessel, the current void concentration will lower due to pressure collapse. When this occurs there is less backpressure (two phase flow restriction) within the core region with no change in recirc pump speed – downcomer water level will initially lower (this is the opposite reactor water level response when a SRV is opened.)

C. Incorrect: Reactor power rise will occur after the core voids have collapsed and are filled with downcomer water inventory.

D. Incorrect: Reactor water level control is level always dominate even in 3 element control. The Steam flow is an anticipatory signal (which is slower to influence the response of the feedwater level control signal.) The lowering of “D” Main Steam line is distributed among the other 3 Main Steam lines. The lowering and rising of steam flows will have negligible effect to reactor water level control.

Technical Reference(s): AOP 262, Rev. 9 (Attach if not previously provided)

BR08Ir3a BT08Ir3a

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.54.00.03, Evaluate the indications provided to predict the expected plant response (for AOP 262)	(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	

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

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295026	EK3.05	
	Importance Rating	3.9		

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor SCRAM

Proposed Question: RO Question 13

- The plant is operating at 95% reactor power
- An SRV is leaking into containment
- Torus cooling is in service
- Torus average water temperature is 800F rising at 10F per minute

With no changes to the given plant conditions and assuming Torus average water temperature continues to rise at 10F per minute, how long does the crew have until a manual reactor scram is required?

A. 15 minutes

B. 25 minutes

C. 30 minutes

D. 40 minutes

Proposed Answer: C

Explanation (Optional):

A. Incorrect: With the conditions provided in the STEM, at this temperature rise of 10F per minute the time would calculated to $15\,min = ((95-80)0F * \frac{min}{1\,0F})$. 950F is identified in Technical Specification 3.6.2.1, Suppression Pool Average Temperature and EOP 2, Primary Containment Control to take action initiate Torus Cooling. This was already provided in the STEM of the question.

B. Incorrect: With the conditions provided in the STEM, at this temperature rise of 10F per minute the time would calculated to $25\,min = ((105-80)0F * \frac{min}{1\,0F})$. 1050F is identified in Technical Specification 3.6.2.1, Suppression Pool Average Temperature as a breakpoint to secure adding heat energy to the Torus.

C. Correct: With the conditions provided in the STEM, at this temperature rise of 10F per minute the time would calculated to $30\,min = ((110-80)0F * \frac{min}{1\,0F})$. 1100F is identified in Technical Specification 3.6.2.1, Suppression Pool Average Temperature and EOP 2, Primary Containment Control. At this temperature, the reactor being shutdown is assumed for the containment loads analyses.

D. Incorrect: With the conditions provided in the STEM, at this temperature rise of 10F per minute the time would calculated to $40\,min = ((120 - 80)0F * \frac{min}{10F})$. 1200F is identified in Technical Specification 3.6.2.1, Suppression Pool Average Temperature as a breakpoint which requires reactor pressure vessel depressurization assumed for the containment loads analyses.

Technical Reference(s): TS 3.6.2.1, Amd. 223 (Attach if not previously provided)

Bases-EOP 2, Rev. 16

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.17, evaluate plant status and control room indications to determine the applicability and affect of any EOP Curve or Limit (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

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EA1.04	
	Importance Rating	3.9	

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell pressure

Proposed Question: RO Question 14

- The plant is operating at full reactor power
- One (1) Torus-to-Drywell vacuum breaker failed to return to its normal position during testing
- An indication problem has been ruled out

If a DBA LOCA occurs in this condition:

- PEAK Drywell pressure will be __(1)__ the peak drywell pressure during a DBA LOCA with all vacuum breakers in their normal position.
- PEAK Drywell temperature will be __(2)__ the peak drywell temperature during a DBA LOCA with all vacuum breakers in their normal position.

	(1)	(2)
A.	the same as	the same as
B.	higher than	the same as
C.	higher than	lower than
D.	higher than	higher than

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Plausible if it is believed that the Drywell to Torus Vacuum Breaker being open during a DBA LOCA will have no different effect on DW temp or pressure that was assumed in the initial design basis accident.

B. Incorrect – Plausible if it is believed that temperature is not affected by a rise in DW pressure.

C. Incorrect – Plausible if it is believed that temperature will be lower during a DBA LOCA with the Drywell to Torus Vacuum Breaker being

D. Correct - A failed open Drywell to Torus vacuum breaker also reduces the pressure suppression function of the containment. The drywell will peak at a higher pressure than if the vacuum breaker is closed, because some steam will bypass the suppression pool

and enter the Torus airspace. Since peak drywell pressure will be higher, peak drywell temperature will be higher (saturation properties of steam)

Technical Reference(s): SD-959, Rev. 4 (Attach if not previously provided)

BT03Ir3a

Proposed References to be provided to applicants during examination: N

Learning Objective: 42.00.00.07, explain, in detail, the Primary Containment response to the design basis (DBA) LOCA, including assumed initial conditions and post-accident equilibrium conditions. (As available)

Question Source:	Bank #	54294	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: PDA 17-1 EOP Comprehensive Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EA1.06	
	Importance Rating	3.4	

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Condensate storage and transfer (make-up to the suppression pool): Plant-Specific

Proposed Question: RO Question 15

As directed by EOP 2 to raise torus water level, which of the following will provide a flow path from the Condensate Storage Tank to the Torus?

A. Gravity drain the CST to the Torus using the Core Spray System

B. Align Condensate and Feed flow through the Torus Spray Header

C. Align Control Rod Drive System flow through the Torus Spray Header

D. Gravity drain the CST to the Torus using the Reactor Core Isolation Cooling System

Proposed Answer: A

Explanation (Optional):

A. Correct – OI 151, Core Spray System, section 11, Raising Torus Level with the Core Spray System, contains a Caution which provides the discussion that during normal operations, CST Suction Valve V-21-1[2] should only be throttled open approximately two to four turns to ensure the valve can be closed quickly when desired Torus level is reached; otherwise, the Torus may overfill during the time it takes the operator to close the valve. V-21-01, Core Spray Pump 1P-211A CST Suction Isolation, is a 10 inch diameter valve which cross connects the CST to the Torus.

B. Incorrect – The Condensate and Feed System take suction from the Hotwell which uses the CST as a source for makeup. This is plausible if the candidate is using another system lineup to complete the alternate injection pathway.

C. Incorrect – The Control Rod Drive System has the ability to take a direct suction from the CST. This is plausible if the candidate believes that there is a cross connection between the two systems.

D. Incorrect – This would be correct if the system selected was the High Pressure Coolant Injection System.

Technical Reference(s): Bases-EOP 2, Rev. 16 (Attach if not previously provided)

OI-151, Rev. 78

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.21 (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK2.14	
	Importance Rating	3.9	

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following:
Emergency generators

Proposed Question: RO Question 16

- The plant is operating in MODE 2

An event occurs with the following timeline:

TIME	Reactor Level (inches)	Reactor Pressure (PSIG)	Drywell Pressure (PSIG)
00:00:00	180	900	0.8
00:05:00	170	850	1.0
00:10:00	119	820	1.3
00:15:00	64	800	1.5
00:20:00	15	750	1.8
00:25:00	-25	720	2.0

No additional Operator action and based on the above timeline, which of the following is correct?

A. A Group 5 Isolation occurred at approximately 00:05:00

B. Main Steam Isolation Valves automatically closed at approximately 00:10:00

C. Both SBDGs automatically started at approximately 00:15:00

D. LPCI Loop Select will have occurred at approximately 00:25:00

Proposed Answer: C

Explanation (Optional):

A. Incorrect: A Group 5 isolation would have occurred at 00:10:00 when reactor water level indicated 119.5 inches.

B. Incorrect: MSIV's will not occur until time 00:15:00 when reactor water level indicates 64 inches.

C. Correct: Both SBDGs will have received an automatic start signal at 00:15:00 when reactor water level indicates 64 inches.

D. Incorrect: LPCI Loop Select will have occurred at 00:10:00 when reactor water level indicated 119.5 inches.

Technical Reference(s): SD-324, Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:			N
Learning Objective:	19.01.01.07, Explain the SBDG starting logic for a manual or automatic start signal, including setpoints.		(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		

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EA1.04	
	Importance Rating	4.5	

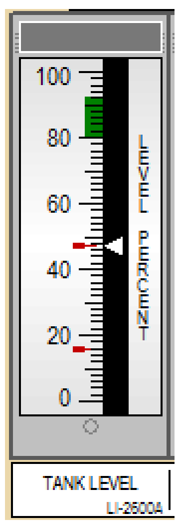
Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Standby Liquid Control System (SBLC)

Proposed Question: RO Question 17

An Anticipated Transient without Scram (ATWS) is in progress with reactor power above the APRM Downscale setpoint and Standby Liquid Control (SBLC) has been initiated.

Sometime later, the Operator observes the following:

- Reactor power is less than the Point of Adding Heat (POAH)
- The following level instrument for SBLC at 1C05:



Which of the following statements are correct for the given plant condition?

A. Cold Shutdown Boron Weight has been injected and a cooldown can commence

B. Hot Shutdown Boron Weight has been injected, system operation is still required

C. SBLC Pumps are required to be secured because the reactor is less than the POAH

D. SBLC Heaters are required to be secured to prevent damage due to being uncovered

Proposed Answer: B

Explanation (Optional):

A. Incorrect: From the picture of the SBLC tank level indication only Hot Shutdown Boron Weight has been injected. Cold Shutdown Boron Weight is the second mark on the gauge face at 16%.

B. Correct: ATWS bases direct that continued operation of SBLC is required until Cold Shutdown Boron Weight has been injected to ensure the reactor is shutdown under all conditions with boron and allow for plant cooldown. From the picture provided, the indicated level of 47% is equivalent to Hot Shutdown Boron Weight injected into the vessel.

C. Incorrect: ATWS bases direct that continued operation of SBLC is required until Cold Shutdown Boron Weight has been injected to ensure the reactor is shutdown under all conditions with boron and allow for plant cooldown. OI 153 QRC 1 directs securing the SBLC Pumps when tank level approaches 0%.

D. Incorrect: OI 153 QRC 1, directs that the SBLC heaters should be secured when tank level indicates about 30% at 1B34 and 1B14.

Technical Reference(s): Bases-ATWS, Rev. 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 6.01.01.02, identify the appropriate procedure that governs the Standby Liquid Control System operation, including operator responsibilities during all modes of operation, and any actions required by personnel outside the Control Room. (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EA2.04	
	Importance Rating	4.1	

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Source of off-site release

Proposed Question: RO Question 18

- A radiological release is in progress

NOTE: Analyze each location separately with the same release concentration rate.

Which of the following sources of the release would pose the greatest radiological exposure to the general public?

A. Turbine Building South rollup door

B. Turbine Building Exhaust

C. Main Plant Exhaust

D. Off-gas stack

Proposed Answer: A

Explanation (Optional):

A. Correct: The bases of EOP 4 have Operators align turbine building and Main Plant ventilation to ensure that a ground level release is minimized.

B. Incorrect: The first and second continuous recheck statements of the EOP 4 flow chart have Operations ensure that the Turbine Building exhaust fan operation is in progress to minimize a ground level release.

C. Incorrect: The second continuous recheck statement of the EOP 4 flow chart have Operations keep Main Plant Exhaust ventilation in service unless there is an indication that the Group 3/SBGT are challenged to minimize the release through this flow path.

D. Incorrect: The SBGT system will discharge a filtered effluent using the Off-gas stack. The use of the Off-gas stack is to aid in the dilution of any radiological effluents that were not filtered by the SBGT system.

Technical Reference(s): Bases-EOP 4, Rev. 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.03, Explain the overall mitigation strategy of the EOPs	(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	

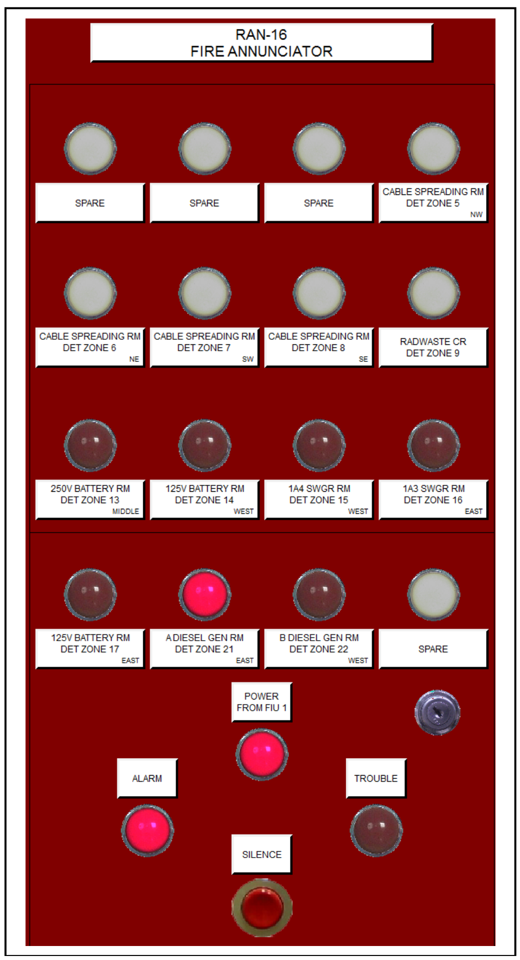
Comments: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 G2.1.20	
	Importance Rating	4.6	

Ability to interpret and execute procedure steps: Plant Fire On Site

Proposed Question: RO Question 19

- An Operator observes the following alarms.



- There are NO additional annunciators at 1C040

In accordance with AOP 913, Fire, the Fire Brigade ___(1)___ required to muster and the pre-action system ____(2)____ activated?

	(1)	(2)
A.	is	has

B.	is	has not
C.	is not	has
D.	is not	has not

Proposed Answer: B

Explanation (Optional):

A. Incorrect: From the information provided in the STEM of the question, “No additional annunciators are observed to be in alarm at 1C040 or on the RAN 16,” indicates that the preaction system has not actuated. Activation would have been followed by the start of the electric or diesel fire pump.

B. Correct: In accordance with AOP 913, FIRE, annunciator A DIESEL GEN. RM. DET. ZONE 21 (EAST) requires immediate fire brigade activation. From the information provided in the STEM of the question, “No additional annunciators are observed to be in alarm at 1C040 or on the RAN 16,” indicates that the preaction system has not actuated. Activation would have been followed by the start of the electric or diesel fire pump. From SD-513, when temperature exceeds the given setpoint, a Heat Activated Device (HAD) or remotely actuated solenoid valve (Turbine Bearing Pre-action System) will vent priming water pressure from the top of the deluge valve main disc. When flow is allowed to pass through the deluge valve, the pressure in the downstream side will open the Pressure Operated Relief Valve (PORV), ensuring deluge system flow until the system isolation valve is closed.

C. Incorrect: This would be true if an Operator is required to verify a presence of a fire. The building in-plant Operator would respond to investigate to determine if a fire is present and to size up the fire.

D. Incorrect: This would be true if an Operator is required to verify a presence of a fire. The building in-plant Operator would respond to investigate to determine if a fire is present and to size up the fire.

Technical Reference(s): AOP 913, Rev. 77 (Attach if not previously provided)

ARP 1C40, Rev. 71

Proposed References to be provided to applicants during examination: N

Learning Objective: 34.05.01.01, Relate the precautions and limitations, operating cautions, or procedural notes of OI-513 to any component or fire protection system operating status (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:	PDA 2015 Audit Exam	Last NRC Exam:	

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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AK1.03	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Under-excitation

Proposed Question: RO Question 20

- The plant is operating at full reactor power
- ITC informs the control room of grid disturbances in southern Iowa

Current Main Generator conditions are:

- 640 MWe Gross
- 50 MVAR Lagging (out)
- 45 psig Hydrogen pressure

ITC Midwest informs the control room that due to the grid disturbances they will be raising surrounding grid voltage.

The Main Generator MVAR indication will change toward the ___(1)___ direction and the ___(2)___ will protect the Main Generator from damage.

A. (1) lagging
(2) over-excitation limiter

B. (1) leading
(2) over-excitation limiter

C. (1) lagging
(2) under-excited reactive ampere limiter (URAL)

D. (1) leading
(2) under-excited reactive ampere limiter (URAL)

Proposed Answer: D

Explanation (Optional):

A. Incorrect: From OI 698 Precaution and Limitation (P&L) 16, lowering grid voltage will operate the Main Generator MVAR loading further from the URAL. This is accomplished by making the Main Generator MVAR loading trend toward the lag direction. If the surrounding grid voltage is raised, this will cause the main generator MVAR loading to trend toward the lead direction, toward the URAL.

B. Incorrect: From OI 698 the over-excitation limiter shows protection if the Main Generator trends towards the lagging direction. Not trending toward the leading direction.

C. Incorrect: From OI 698 the Under-excited reactive ampere limiter (URAL) shows protection if the Main Generator trends towards the leading direction. Not trending toward the lagging direction.

D. Correct: From OI 698 Precaution and Limitation (P&L) 16, lowering grid voltage will operate the Main Generator MVAR loading further from the URAL. This is accomplished by making the Main Generator MVAR loading trend toward the lag direction. If the surrounding grid voltage is raised, this will cause the main generator MVAR loading to trend toward the lead direction, toward the URAL. From OI 698 the Under-excited reactive ampere limiter (URAL) shows protection if the Main Generator trends towards the leading direction.

Technical Reference(s): OI-698, Rev. 97 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 57.00.00.04, analyze the estimated capabilities for limiting conditions and operating parameters tolerances (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295007 AK3.06	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Reactor/turbine pressure regulating system operation

Proposed Question: RO Question 21

- The plant is operating at 56% reactor power
- An Electro-Hydraulic Control (EHC) logic failure has enforced a false stator cooling water runback signal
- Main Generator electrical output is observed to be lowering

No Operator actions are taken, reactor pressure will _______.

A. remain unaffected due to current reactor power

B. continually rise and cause an automatic reactor scram

C. rise and stabilize at 5 psig higher when the runback completes

D. lower due to bypass valves opening to reduce generator electrical output

Proposed Answer: B

Explanation (Optional):

A. Incorrect: A Main Generator runback signal will cause the generator to lower and eventually trip on reverse power. At the provided reactor power (>26%) the turbine bypass and auxiliary steam loads will not use the remaining steam generation which will cause reactor pressure to continuously rise.

B. Correct: For the given reactor power and the input of a stator cooling water runback; this combination would cause reactor pressure to rise and RPS will receive either a scram signal on high neutron flux or reactor pressure.

C. Incorrect: The fault provided in the STEM will cause a generator load reduction to a generator output of 25%. Turbine bypass valve capabilities and auxiliary steam loads can sink 26% steam demand. This would lower the generator percentage to 30%, with a difference of 5% between current and the stator cooling runback of 25%. In addition, this is a typical response to a pressure regulator failure downscale (lowering pressure.)

D. Incorrect: The bypass control valves will come open, but the reason for this is not to lower generator electrical output. The lowering of generator electrical output by closing the turbine control valves will cause bypass valves to open to maintain equalizing manifold pressure at setpoint.

Technical Reference(s): SD-693-2A, Rev. 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	52.00.00.05, describe the operation of the following principle EHC Logic System components: a. pressure control unit, b. bypass control unit, c. speed and acceleration control unit, d. load control unit, and e. valve positioning unit	(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	

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009 AK1.05	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: Natural circulation

Proposed Question: RO Question 22

- The plant was operating at full reactor power
- The crew inserted a manual reactor scram
- Both recirculation pumps have tripped
- Reactor water level was recovered to 190 inches and is stable
- Reactor pressure was lowered to 550 psig and is stable
- The reactor scram is reset

Which of the following strategies listed below is directed by AOP 264, Loss of Recirc Pump(s), to aid in core cooling?

A. Secure Reactor Building Closed Cooling Water to limit recirc loop cooldown and prevent thermal stratification

B. Secure Reactor Water Cleanup to limit reactor vessel bottom head cooldown and prevent thermal stratification

C. Raise Control Rod Drive system flow using FI-1814, CRD System Flow Control, to force water flow though the core region

D. Raise reactor water level with condensate pumps to 230 to 240 inches to promote natural circulation through the core region

Proposed Answer: D

Explanation (Optional):

A. Incorrect: AOP 264 directs that mini-purge is to be secured to recirc pump seals to prevent thermal stratification.

B. Incorrect: AOP 264 directs using Reactor Water Cleanup (either by operating the pumps or using the system for drain) to prevent thermal stratification.

C. Incorrect: AOP 264 directs Operators to lower Control Rod Drive system flow to approximately 10 gpm to prevent introducing cold water into the reactor vessel bottom head and causing thermal stratification.

D. Correct: From the conditions provided with no recirc pumps, AOP 264 directs that the crew should raise reactor water level to approximately 230-240 inches to improve natural circulation. With the provided reactor pressure, the condensate pumps can perform this action.

Technical Reference(s): AOP 264, Rev. 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 94.58.00.04, relate how each step and its performance meets the mitigation strategies of AOP 264 (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 8

55.43

Comments: Components, capacity, and functions of emergency systems.

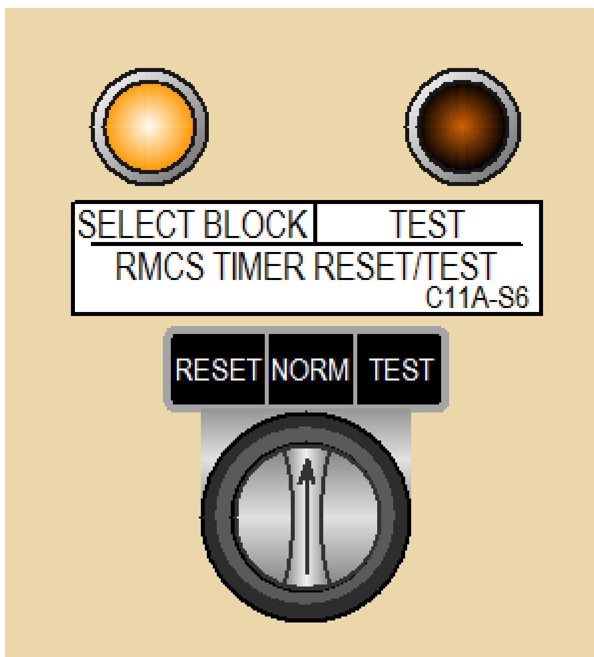
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014 AK2.08	
	Importance Rating	3.4	

Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: Reactor Manual Control System (RMCS) Plant-Specific

Proposed Question: RO Question 23

- The plant is operating at 95% reactor power
- The Operator is moving control rods for a sequence exchange
- The Operator observes the selected control rod has “bailed”

The following was observed on 1C05:



Which of the following would have produced this indication?

A. An in-sequence control rod is given a continuous insert signal using the ROD MOVEMENT CONTROL switch

B. Using the EMER ROD IN position of the EMERG IN/NOTCH OVERRIDE select switch to continuously insert a control rod

C. An in-sequence control rod is given a single notch withdrawal signal and the red ROD OUT light remains energized for 3 seconds during control rod movement

D. Holding the ROD MOVEMENT CONTROL switch in the NOTCH OUT position WITHOUT also operating the EMER ROD IN/NOTCH OVERRIDE select switch

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The RWM will insert a RWM ROD BLOCK.

B. Incorrect: Using EMER IN interrupts the timer, but does not cause a SELECT BLOCK.

C. Correct: The TIMER MALFUNCTION SELECT BLOCK will activate if the control rod is given a withdrawal signal for more than two seconds. This block is bypassed when doing an intentional continuous withdrawal.

D. Incorrect: Withdrawal will be one notch only, but this is a time function, not a select block.

Technical Reference(s): SD-856-1, Rev. 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 72.00.00.02, describe the operation of the following principle Reactor Manual Control System components: b Rod Select Relays (As available)

Question Source: Bank # 22354

Modified Bank # (Note changes or attach parent)

New

Question History: PDA 17-1 Systems Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295015	AA1.02	
	Importance Rating	4.0		

Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Reactor Protection System (RPS)

Proposed Question: RO Question 24

- The plant is operating at full reactor power
- The crew inserts a manual reactor scram

The following was observed at 1C05:

With these indications the Operator at the Controls will first _______.

A. insert IRMs and SRMs

B. inject Standby Liquid Control

C. lower recirc speed to minimum

D. lower the Master FRV controller to 158.5 inches

Proposed Answer: D

Explanation (Optional):

A. Incorrect – the Operator will not insert IRMs and SRMs until reactor power has been lowered. This action would be performed for a successful reactor scram or a low power ATWS condition.

B. Incorrect – The Operator will report the status of the nuclear parameters prior to injecting Standby Liquid Control.

C. Incorrect – The Operator will verify both recirc pumps have lowered to 20% when the feedwater flow rate is lowered.

D. Correct – The Operator will lower reactor water level immediately in preparation to lower reactor power through RPV water level reduction.

Technical Reference(s): IPOI 5 QRC 2, Rev. 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.55.01.01, Explain how the mitigation strategies used in ATWS accomplish the purpose of ATWS	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295017 AK3.01	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System isolations

Proposed Question: RO Question 25

- The plant is operating at full reactor power
- The crew inserts a manual reactor scram
- Both SBGT trains start
- A complete Group 3 isolation occurs

The following indications are then noted:

- 1C35A(C-3), Reactor Building KAMAN 3, 4, 5, 6, 7 & 8 RAD Trouble, is in alarm
- The alarm is determined to be due to rising radiation levels on the KAMAN monitors

In accordance with ARP 1C35A(C-3) and OI 170, Standby Gas Treatment System, which of the following identifies the earliest point where operator action is required?

When the KAMAN monitors reach the ______.

A. High-High (red) setpoint, ALL running Main Plant Exhaust Fans must be secured

B. High level (yellow) setpoint, ALL running Main Plant Exhaust Fans must be secured

C. High-High (red) setpoint, the number of running Main Plant Exhaust Fans must be reduced down to ONE fan

D. High level (yellow) setpoint, the number of running Main Plant Exhaust Fans must be reduced down to ONE fan

Proposed Answer: A

Explanation (Optional):

A. Correct: IAW ARP IC35A C-3 and OI 170 precaution #10, Main Plant Exhaust Fans 1V-EF-1, 1V-EF-2, and 1V-EF-3 have to be shutdown if SBGT A[B] is running due to a Group III isolation signal and Reactor Building KAMAN red alarm condition exists. The "red" alarm corresponds to the High-High setpoint. This is done to prevent bypass of the SGTS filter units by air from the reactor building via the main plant ventilation stack and precludes or limits an untreated release to the environs.

B. Incorrect: This action is required at the High-High setpoint.

C. Incorrect: All fans must be secured. Plausible if the operator believes that reducing the number of running fans would reduce the total amount of radioactivity released.

D. Incorrect: Action is not required to the High-High setpoint is reached. Additionally, the required action is to secure all fans.

Technical Reference(s): ARP 1C35A, Rev. 37 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 87.01.01.01, (As available)

Question Source: Bank # 47017

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: PDA 2013

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EA2.01	
	Importance Rating	3.9	

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Suppression pool water level

Proposed Question: RO Question 26

Following a plant transient and a manual reactor scram, the following containment parameters are observed:

- Drywell air temperature is below the Drywell Spray Initiation Limit
- Torus water level is 14.5 feet and stable

What would be the effect of initiating Torus and Drywell Sprays under these conditions?

Drywell pressure ________.

A. will lower rapidly and with the Torus to Drywell vacuum breakers submerged, this will prevent Nitrogen return to the Drywell and result in containment failure

B. will lower rapidly which will exceed the capacity of the Reactor Building to Torus Vacuum Breakers and result in containment failure

C. and Torus pressure will slowly lower as Nitrogen is returned from the Torus to the Drywell and result in no containment challenges

D. and Torus pressure will slowly lower with the lower Spray Ring Header submerged and result in no containment challenges

Proposed Answer: A

Explanation (Optional):

A. Correct: The Torus to Drywell vacuum breakers will begin to submerge at 13.5 feet in the Torus. With these vacuum breakers being submerged, Nitrogen will be impeded to return the Drywell. Negative containment pressure or torus-to-drywell dP from drywell sprays may cause drywell or torus failure.

B. Incorrect: The concern with the lowering pressure is that the Torus to Drywell vacuum breakers will not operate as designed. If these vacuum breakers fail to function as designed the negative pressure is not caused by in the inability of the Reactor Building to Torus vacuum breakers to operate.

C. Incorrect: This would be correct if the Torus to Drywell vacuum breakers were not submerged.

D. Incorrect: This would be correct if the Torus to Drywell vacuum breakers were not submerged.

Technical Reference(s):	Bases-Breakpoints, Rev. 14		(Attach if not previously provided)
Bases-EOP 2, Rev. 16			

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.59.03.01,		(As available)
Question Source:	Bank #	47310	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	PDA 17-1 Workup Comp	Last NRC Exam:	

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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032	G2.1.30
	Importance Rating	4.4	

Ability to locate and operate components, including local controls: High Secondary Containment Area Temperature

Proposed Question: RO Question 27

A primary leak from Reactor Water Cleanup (RWCU) is in progress in the RWCU Heat Exchanger (HX) Room.

- RWCU HX Room temperature is 2200F and rising
- MO-2700, RWCU Inlet Inboard Isolation valve will not close
- MO-2701, RWCU Suction Outboard Isolation valve <u>cannot</u> be operated from 1C04

Is this leak isolable by local operation?

A. No, MO-2701 is located in the Drywell

B. No, MO-2701 is located in the RWCU HX Room

C. Yes, MO-2701 is located in the Steam Tunnel

D. Yes, MO-2701 is located in area above tip room

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This would be correct if the question was asking about MO-2700.

B. Correct: MO-2701 is located in the RWCU HX Room and is normally accessible during plant operation. The room temperature is 2200F and this temperature is greater than MAX Safe, as indicated on EOP Table 6. At this temperature, the valve will be inaccessible for local operation.

C. Incorrect: This would be correct if MO-2701 was located in the Steam Tunnel.

D. Incorrect: This would be correct if MO-2701 was located in the area above the tip room. MO-2740, is located at this location.

Technical Reference(s): Bases-EOP 3, Rev. 13 (Attach if not previously provided)

SD-261, Rev. 9

Proposed References to be provided to applicants during examination: EOP Table 6

Learning Objective:	11.00.00.03, describe the function/operation of the following principle RWCU system components: c. RWCU Motor Operated Valves	(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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 K2.01	
	Importance Rating	3.5	

Knowledge of electrical power supplies to the following: Pumps

Proposed Question: RO Question 28

- The plant has a complete loss of off site power
- 1G-31, A Standby Diesel Generator, failed

Which of the following identifies the RHR pumps which will be available?

	1P-229A	1P-229B	1P-229C	1P-229D
A.	available	-	-	available
B.	available	-	available	-
C.	-	available	available	-
D.	-	available	-	available

Proposed Answer: D

Explanation (Optional):

A. Incorrect: 1P-229A is powered from 1G-31, "A" Standby Diesel Generator. From the conditions provided, 1A3 has no electrical power supplied due to failure of the SBDG and a loss of offsite power.

B. Incorrect: 1P-229A and 1P-229C are powered from 1G-31, "A" Standby Diesel Generator. From the conditions provided, 1A3 has no electrical power supplied due to failure of the SBDG and a loss of offsite power.

C. Incorrect: 1P-229C is powered from 1G-31, "A" Standby Diesel Generator. From the conditions provided, 1A3 has no electrical power supplied due to failure of the SBDG and a loss of offsite power.

D. Correct: With a loss of offsite electrical power 1G-21, B Standby Diesel Generator would have started and powered 1A4. 1A4 provides electrical power to 1P-229B and 1P-229D.

Technical Reference(s): SD-149, Rev. 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	2.01.01.06, Given an RHR system operating mode and various plant conditions, predict how the RHR system will be impacted by operation, or failure of the following support systems: a. Essential 4160/480 VAC electrical power supplies	(As available)

Question Source:	Bank #	
	Modified Bank # 46088	(Note changes or attach parent)
	New	

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Modified bank question by changing which SBDG failed to Operate. This made answer choice D correct and answer choice B incorrect.

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 29

- The plant is in MODE 4
- The “A” RHR and the “A” RHRSW pumps running for shutdown cooling (SDC)
- SDC is aligned to the “A” recirculation loop
- Alternate RPS is unavailable due to testing
- 1B301, CB 480VAC Motor Control Center 1B32, trips on an overload

What will be the impact on SDC?

A. “A” RHR pump will trip

B. “A” RHR pump will run “dead-headed”

C. “A” RHR pump will continue to run on minimum flow

D. “A” RHR pump will continue to run with flow unchanged

Proposed Answer: A

Explanation (Optional):

A. Correct: A loss of 1B301 will cause 1B32 to de-energize and causing a loss of “A” RPS. This will result in MO-1908, inboard SDC valve, to go shut and tripping the “A” RHR pump on a loss of suction path.

B. Incorrect: A loss of 1B301 will cause 1B32 to de-energize and causing a loss of “A” RPS. This will result in MO-1908, inboard SDC valve, to go shut and tripping the “A” RHR pump on a loss of suction path.

C. Incorrect: A loss of 1B301 will cause 1B32 to de-energize and causing a loss of “A” RPS. This will result in MO-1908, inboard SDC valve, to go shut and tripping the “A” RHR pump on a loss of suction path.

D. Incorrect: A loss of 1B301 will cause 1B32 to de-energize and causing a loss of “A” RPS. This will result in MO-1908, inboard SDC valve, to go shut and tripping the “A” RHR pump on a loss of suction path.

Technical Reference(s): AOP 358, Rev. 32 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:	N	
Learning Objective:	50007.05.11, Predict the plant response to a group isolation from PCIS signal inputs and plant conditions	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 G2.4.50	
	Importance Rating	4.2	

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual: High Pressure Coolant Injection System

Proposed Question: RO Question 30

A plant transient has occurred which required the crew to insert a manual reactor scram.

- Reactor water level is 100 inches and lowering
- Drywell pressure is 1.4 psig and stable

The Balance of Plant (BOP) Operator observes the following at 1C03:

- High Pressure Coolant Injection (HPCI) is in a standby readiness condition

Is this expected for the given plant conditions, and if not, what actions are required?

A. Yes, HPCI will receive an auto start signal when 1C05A (A-1), Reactor LO-LO-LO Level Trip, annunciator is received and no further action is required

B. Yes, HPCI will receive an auto start signal when 1C05B (A-1), Primary Containment HI Pressure Trip, annunciator is received and no further action is required

C. No, HPCI should have received an auto start signal when 1C05A (C-1), Reactor LO Level Trip, annunciator was received and the BOP should manually start HPCI

D. No, HPCI should have received an auto start signal when 1C05A (B-1), Reactor LO-LO Level Trip, annunciator was received and the BOP should manually start HPCI

Proposed Answer: D

Explanation (Optional):

A. Incorrect: This would be correct if the question asked when LPCI auto initiated.

B. Incorrect: This is not expected for the give plant conditions. HPCI should have automatically started when reactor water level lowered to Reactor LO-LO Level Trip. This annunciator is an auto start signal, however drywell pressure is < 2 psig.

C. Incorrect: This would be correct if the question asked for an automatic reactor scram signal and a scram was not received.

D. Correct: HPCI should have automatically started when reactor water level lowered to Reactor LO-LO Level Trip setpoint of 119.5 inches. The Operator is required to manually start HPCI.

Technical Reference(s):	ARP 1C05A, Rev. 83	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	5.02.01.02, List the signals which cause a HPCI System auto initiation including setpoints and logic.	(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	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A1.04	
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Reactor pressure

Proposed Question: RO Question 31

An event has occurred and an RPV Emergency Depressurization is in progress.

- All Low Pressure Emergency Core Cooling Systems (ECCS) pumps are running on minimum flow
- Reactor pressure is 600 psig and lowering

In accordance with EOP 1, RPV Control, below which reactor pressure will the Balance of Plant Operator first observe Core Spray flow indication?

A. 450 psig

B. 330 psig

C. 260 psig

D. 135 psig

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This is the pressure in the reactor vessel which the low pressure Emergency Core Cooling Systems (ECCS) inject valves will receive an OPEN signal. At this pressure, each low pressure ECCS pumps will not have sufficient pressure to cause indicated flow at 1C03.

B. Correct: SD-151 and EOP 1 identify that the nominal shutoff head for Core Spray is 330 psig. As reactor pressure is lowered to less than 330 psig, but before 260 psig, Core Spray flow will be observed at the 1C03 panel.

C. Incorrect: At this pressure Core Spray will be injecting at designed flow rates, however this is not the pressure which the Balance of Plant Operator will first observe Core Spray flow indication. This is the pressure identified for Low Pressure Coolant Injection (LPCI) System will demonstrate system flow into the vessel.

D. Incorrect: At this pressure Core Spray will be injecting at designed flow rates, however this is not the pressure which the Balance of Plant Operator will first observe Core Spray flow indication. This is the pressure which the Shutdown Cooling Interlocks will clear.

Technical Reference(s):	Bases-EOP 1, Rev. 18	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	4.01.01.02, evaluate plant conditions and control room indications to determine if the Core Spray system is operating as expected, and identify any actions that may be necessary to place the Core Spray system in the correct lineup.	(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	

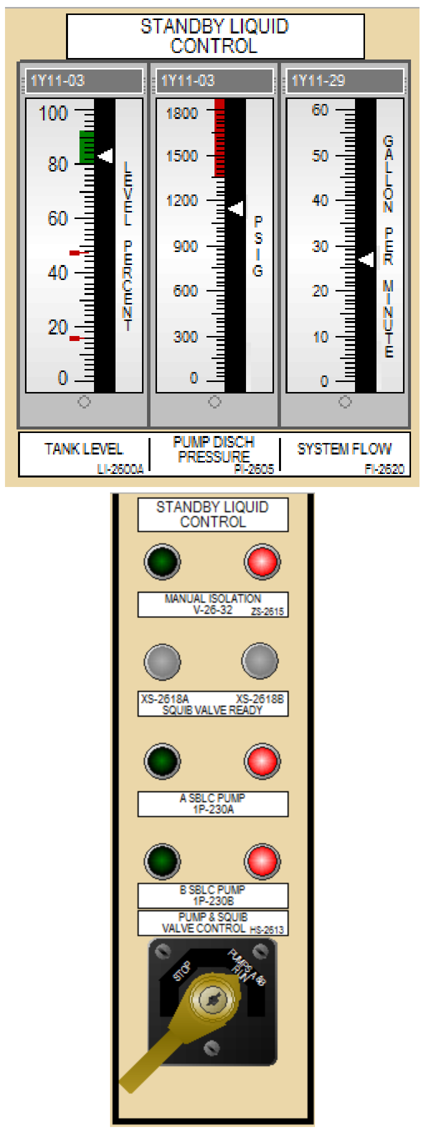
Comments: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K3.01	
	Importance Rating	4.3	

Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: Ability to shutdown the reactor in certain conditions

Proposed Question: RO Question 32

- The crew initiates Standby Liquid Control (SBLC) with the following indications:



What is the status of SBLC and its ability to inject Hot Shutdown Boron Weight (HSBW)?

A. SBLC will inject HSBW within its normal time

B. SBLC will inject HSBW in less than its normal time

C. SBLC will inject HSBW in more than its normal time

D. SBLC is currently NOT injecting into the vessel

Proposed Answer: C

Explanation (Optional):

A. Incorrect: From the demonstrated flow rate, one of the SBLC pumps is injecting at designed flow rate. This flow rate is less than normal 2 pump operation.

B. Incorrect: The demonstrated system flow rate indicates that only one pump (or equivalent flow of one pump) is charging into the reactor vessel. With the indication of two pumps operating (A and B SBLC pumps) it could be assumed that it will only take half the designed time.

C. Correct: The demonstrated system flow rate indicates that only one pump (or equivalent flow of one pump) is charging into the reactor vessel. With the indication of only one pump injecting, it will take approximately twice as long then normal to inject SBLC.

D. Incorrect: From the picture, it is demonstrated that system flow is equivalent to one SBLC pump with sufficient discharge pressure. If it is misunderstood what reactor pressure is normally at full reactor power (1020 psig) the demonstrated discharge pressure of the system could be assumed to be lower than vessel pressure and the indicated flow would be system flow operating on recirculation to the tank.

Technical Reference(s): SD-153, Rev. 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 6.00.00.06, Evaluate plant conditions and control room indications to determine if the SBLC System is operating as expected, and identify any actions that may be necessary to place the SBLC in the correct lineup (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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.02	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements

Proposed Question: RO Question 33

A malfunction has occurred with the Reactor Protection System (RPS) Channel A1 Level instrument such that it will NOT generate an RPS trip signal at its RPV Low Level Trip setpoint.

If actual reactor water level lowers below the RPV Low Level Trip setpoint, RPS will generate _________.

A. a full SCRAM due to both channels of RPS energizing the SCRAM solenoids

B. a full SCRAM due to both channels of RPS de-energizing the SCRAM solenoids

C. only a half SCRAM due to only one channel of RPS energizing the SCRAM solenoids

D. only a half SCRAM due to only one channel of RPS de-energizing the SCRAM solenoids

Proposed Answer: B

Explanation (Optional):

A. Incorrect: A trip of any one channel in “A” and “B” side (i.e. channel A1 or A2 and B1 or B2) of RPS will cause a full SCRAM by de-energizing the SCRAM. The candidate will select this answer choice if they have the misunderstanding that RPS has only one channel and is required to energize the SCRAM solenoids to actuate (ECCS logic.)

B. Correct: A full SCRAM will be received if the trip of any one channel in “A” and “B” side (i.e. channel A1 or A2 and B1 or B2) of RPS is experienced. The Reactor Protection System performs this action by de-energizing the SCRAM solenoids to SCRAM the control rods.

C. Incorrect: A trip of any one channel in “A” and “B” side (i.e. channel A1 or A2 and B1 or B2) of RPS will cause a full SCRAM by de-energizing the SCRAM. The candidate will select this answer choice if they have the misunderstanding that RPS has only one channel per trips system and is required to energize the SCRAM solenoids to actuate (ECCS logic.)

D. Incorrect: A trip of any one channel in “A” and “B” side (i.e. channel A1 or A2 and B1 or B2) of RPS will cause a full SCRAM by de-energizing the SCRAM. The candidate will select this answer choice if they have the misunderstanding that RPS has only one channel per trips system.

Technical Reference(s):	SD-358, Rev. 9	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	22.00.00.03, list the signals which cause a Reactor Protection System trip including setpoints and logic, and describe how they are bypassed and how they are reset	(As available)

Question Source:	Bank #	54313	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	PDA 17-1 EOP Comprehensive	Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments: 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.

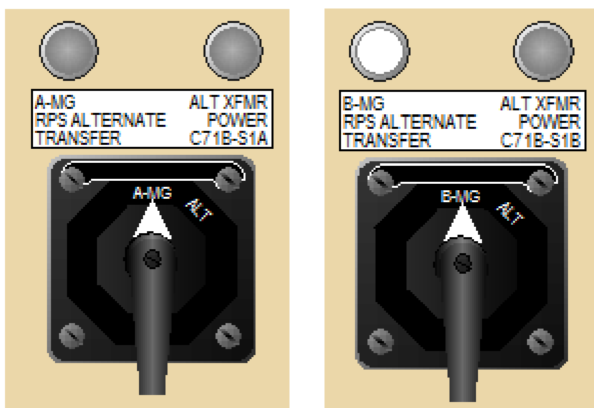
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 A2.02	
	Importance Rating	3.7	

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS bus power supply failure

Proposed Question: RO Question 34

The plant is operating at full reactor power when 1B32 supply breaker tripped open.

- AOP 358, Loss of RPS AC Power, is in progress
- An Operator observes the following indications at 1C15 and 1C17, respectively:



The above indications are expected for RPS which ______________.

A. requires a manual alignment of alternate power to 1B42. The Operator has to reset EPA C1 and C2 and align C71B-S1A to ALT XFMR POWER

B. has alternate power automatically aligned to 1B42 from the power loss. The Operator has to have the EPA A1 and A2 reset and align C71B-S1A to ALT XFMR POWER

C. has a fault on the alternate power supply due to the ALT XFMR POWER light being de-energized with the B-MG indicating light being energized. “A” RPS restoration is not possible

D. requires manual alignment of C71B-S1A to ALT XFMR POWER to energize “A” RPS. The white ALT XFMR POWER indicating light will illuminate to indicate that “A” RPS power has been restored

Proposed Answer: A

Explanation (Optional):

A. Correct: The system alignment of alternate RPS can be aligned to either 1B32 or 1B42. With the loss of 1B32, (ALT XFMR POWER light being de-energized,) must be manually realigned to 1B42 and EPA C1 and C2 reset. Then the transfer can continue.

B. Incorrect: This would be correct if the candidate has the misconception that there is an auctioneering circuit (as Instrument AC and Uninterruptible AC does) that would only require to reset the primary EPA A1 and A2 to restore electrical power.

C. Incorrect: This would be correct if the candidate had the misconception that ALT XFMR POWER light provides the indication that a fault does exist. This picture provides the indication that alternate RPS is de-energized.

D. Incorrect: This would be correct if the candidate had the misconception that the ALT XFMER POWER indicating light would only energize if aligned to alternate power.

Technical Reference(s): SD-358, Rev. 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 94.11.01.03, evaluate plant conditions and control room indications to determine the required actions (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

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K3.01	
	Importance Rating	3.9	

Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: RPS

Proposed Question: RO Question 35

The crew is performing a normal shutdown with the following conditions:

- Reactor power is 12% and stable
- MODE Switch is in RUN
- IRM "D" detector fails UPSCALE while being inserted into the core
- All other IRM's are inserted and on RANGE 10

With the given plant conditions and IRM failure, if the MODE Switch were placed in Start & Hot STBY, which of the following correctly describes the result?

A. Only a half scram will occur once the IRM detectors are fully inserted

B. Only a half scram will occur when the MODE switch is repositioned

C. A half scram and rod block will occur once the IRM detectors are fully inserted

D. A half scram and rod block will occur when the MODE switch is repositioned

Proposed Answer: D

Explanation (Optional):

A. Incorrect: This would be correct if the IRM position was an input to the trip logic.

B. Incorrect: This would be true if a ROD BLOCK was not enforced when the "D" IRM produced an UPSCALE signal

C. Incorrect: This would be correct if the IRM position was an input to the trip logic.

D. Correct: With the "D" IRM producing an UPSCALE signal, when the MODE Switch is moved to Start & Hot STBY, this signal is now un-bypassed in the RPS logic. With a upscale signal, a ROD BLOCK as well.

Technical Reference(s): SD-878-2, Rev. 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	79.00.00.06, Given an IRM System Operating Mode and various plant conditions, predict how each supported system will be impacted by failures in the IRM System: B. RPS	(As available)

Question Source:	Bank #	50907	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A2.01	
	Importance Rating	2.8	

Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power Supply Degraded

Proposed Question: RO Question 36

- A reactor startup is in progress
- The reactor is critical

The following parameters have been established and observed:

- A constant 150 second period has been calculated
- ALL IRMs are on range 4 and reading 50/125

Approximately 30 seconds later, the following indications are observed:

- 1C05A (A-2), "A" RPS AUTO SCRAM alarms
- 1C05A (A-5), NEUTRON MONITORING SYSTEM TRIP
- 1C05A (B-3), IRM "A", "C", OR "E" UPSCALE TRIP OR INOP alarms
- 1C05B (A-6), ROD OUT BLOCK alarm
- "E" IRM Recorder indicates DOWNSCALE

What action will the Operator have to perform first to continue the startup?

A. Bypass IRM "E"

B. Withdraw IRM "E"

C. Range up IRM "E"

D. Reset the half scram

Proposed Answer: A

Explanation (Optional):

A. Correct: The IRM "E" Power supply had to fail downscale as indicated by 1C05A (B-3), IRM "A", "C", OR "E" UPSCALE TRIP OR INOP alarms and "E" IRM Recorder indicates DOWNSCALE. The Operator will have to bypass this IRM to continue the startup in accordance with OI 878.2, Intermediate Range Monitoring System.

B. Incorrect: The IRM will be withdrawn when reactor power is indicating on the power range nuclear instruments and the MODE Switch has been placed in RUN.

C. Incorrect: Would be correct if it was due to the IRM reading greater than 120/125 scale. This would give the 1C05A (B-3) alarm and half scram but not coincident with the IRM recorder being downscale. This provides the indication that the IRM power supply has failed.

D. Incorrect: The half scram will be required to be reset to clear the rod out block, but the condition which caused the rod out block (IRM INOP) must be corrected first.

Technical Reference(s): ARP 1C05A, Rev 83 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 79.00.00.02, evaluate plant conditions and control room indications to determine if the IRM System is operating as expected, and identify any actions that may be necessary to place the IRM System in the correct lineup. (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

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 G2.2.40	
	Importance Rating	3.4	

Ability to apply Technical Specifications for a system: Source Range Monitor (SRM) System

Proposed Question: RO Question 37

A reactor startup is in progress with the following conditions present:

- The MODE Switch is in the START & HOT STBY position
- The crew is preparing to withdraw the first control rod
- All SRM detectors are fully inserted

Below are two possible sets of SRM readings in counts per second (cps).

SET 1 SRM Counts

A	B	C	D
INOP	4	4	3

SET 2 SRM Counts

A	B	C	D
4	1	3	2

For each SET, are the minimum number of required SRM channels available?

	SET 1	SET 2
A.	Yes	Yes
B.	Yes	No
C.	No	Yes
D.	No	No

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This would be correct if the SRM's only required 2 CPS or more to be considered OPERABLE.

B. Correct: L3.3.1.2 and Table 3.3.1.2 require that there are three channels of SRM Operable in the current plant operating mode. With the operable channels indicating ≥3 cps assures that any transient , should it occur, begins at or above the initial value of 10^{-8} of RTP which is used in the analysis of transients in cold conditions.

C. Incorrect: This would be correct if the SRM's only required 4 CPS or more to be considered OPERABLE.

D. Incorrect: This would be correct if the SRM's only required 2 CPS to be considered OPERABLE.

Technical Reference(s):	SD-878-1, Rev. 7	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	78.02.01.07, state when the SRM System is required to be operable by Technical Specifications and describe the bases of the SRM System LCOs	(As available)

Question Source:	Bank #	22668	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: 08-01 ILT Systems Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

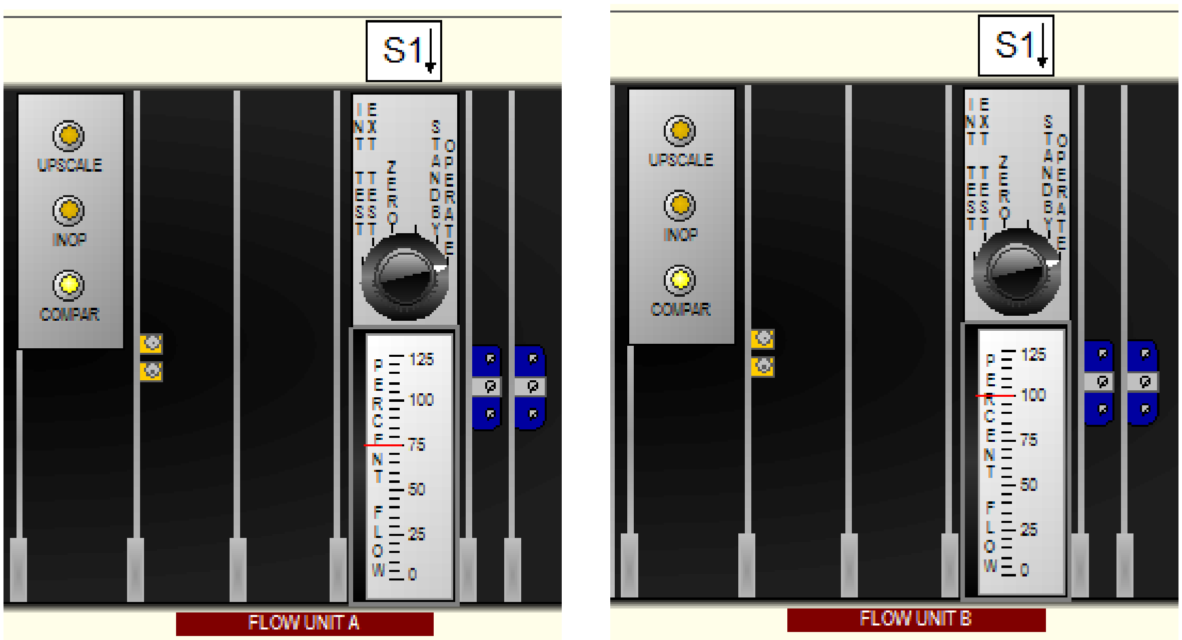
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 K6.07	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Flow converter/comparator network: Plant-Specific

Proposed Question: RO Question 38

- The plant is operating at full reactor power
- Annunciator 1C05A (E-2), APRM Flow Unit Upscale, INOP or Compare Error is received

The following indications are observed at 1C37:



The ___(1)____ flow unit has failed and positioning S1 to ____(2)____ will remove this failed flow signal from the APRM logic.

	(1)	(2)
A.	A	ZERO
B.	A	STANDBY
C.	B	ZERO
D.	B	STANDBY

Proposed Answer: A

Explanation (Optional):

A. Correct: The indicated malfunction of the for the given reactor power is that “A” flow unit has failed downscale (>10%.) To bypass this erroneous signal the Operator will have to place S1 to a position that is not “STANDBY” or “OPERATE.”

B. Incorrect: S1 must be placed in a position other than “STANDBY” or “OPERATE.”

C. Incorrect: “B” flow unit is indicating the correct flow for the given plant condition.

D. Incorrect: “B” flow unit is indicating the correct flow for the given plant condition. S1 must be placed in a position other than “STANDBY” or “OPERATE.”

Technical Reference(s): ARP 1C05A, Rev. 83 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 81.01.01.06, Given an APRM system operating mode and various plant conditions, predict how the APRM system will be impacted by the operation or failure of the following support systems or components: e. APRM Flow Units (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

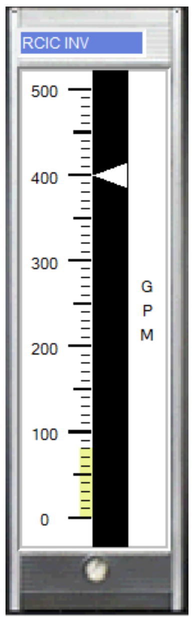
Comments: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A4.08	
	Importance Rating	3.7	

Ability to manually operate and/or monitor in the control room: Reactor Core Isolation Cooling (RCIC) System flow

Proposed Question: RO Question 39

- The plant is operating at 98% reactor power
- RCIC operating in CST-CST mode using the Test Potentiometer (test pot)
- RCIC flow indicates the following



- The crew inserts a manual reactor scram
- Reactor water level lowered to 145 inches and rising
- HPCI automatically starts on a high Drywell pressure

As RCIC continues to operate, the system's flow control will be on the ___(1)___ and the indicated flow will be ___(2)___ flow rate shown above.

	(1)	(2)
A.	test pot	the same
B.	test pot	lower than the
C.	flow controller	the same
D.	flow controller	greater than the

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Upon the receipt of a 2# drywell pressure signal, MO-2316 will close and RCIC minimum flow will open and discharge to the Torus. The minimum flow line is upstream of this flow indicator and the indicated flow will be lower than 400 gpm.

B. Correct: Since reactor water level is maintained above the injection initiation setpoint of 119.5 inches, RCIC will remain controlled by the test pot and not the FIC. Upon the receipt of a 2# drywell pressure signal, MO-2316 will close and RCIC minimum flow will open and discharge to the Torus. The minimum flow line is upstream of this flow indicator and the indicated flow will be lower than 400 gpm.

C. Incorrect: Since reactor water level is maintained above the injection initiation setpoint of 119.5 inches, RCIC will remain controlled by the test pot and not the FIC.

D. Incorrect: Since reactor water level is maintained above the injection initiation setpoint of 119.5 inches, RCIC will remain controlled by the test pot and not the FIC.

Technical Reference(s): SD-150, Rev. 9 (Attach if not previously provided)

SD-152, Rev. 15

Proposed References to be provided to applicants during examination: N

Learning Objective: 3.08.01.04, Describe the RCIC System interlocks, including purpose, setpoints, logic, and when/how they are bypassed (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

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

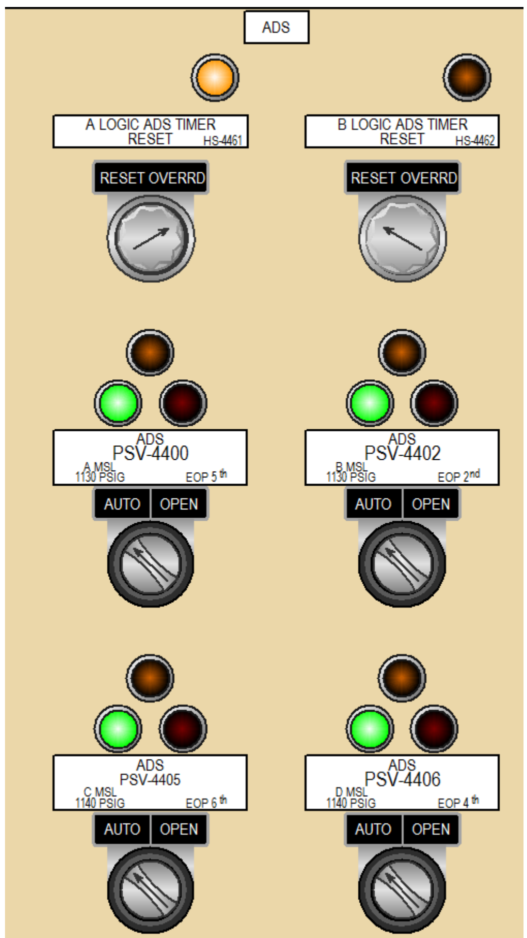
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K5.01	
	Importance Rating	3.8	

Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation

Proposed Question: RO Question 40

- Loss of Coolant Accident occurred
- Bus 1A3, 4160 VAC Essential Switchgear, has a LOCKOUT
- All High Pressure injection sources are unavailable
- Main Steam Isolation Valves are closed
- Reactor water level is 100 inches and lowering at 20 inches per minute

The Operator performs the following when directed to lockout ADS:



- No further Operator action is performed

How will ADS respond in four (4) minutes:

A. All ADS valves will be OPEN

B. All ADS valves remain closed

C. Only PSV-4400 and PSV-4405 will OPEN

D. Only PSV-4402 and PSV-4406 will OPEN

Proposed Answer: A

Explanation (Optional):

A. Correct: At approximately t= 1 min and 48 seconds the “B” logic ADS timer will start and begin the 2 minute count down, as well as, all available low pressure ECCS starts on the 1A4, “B” side. At t=3 min and 48 seconds, the timers will complete the delay time and will initiate ADS will all 4 ADS valves.

B. Incorrect: This would be correct if “B” side ECCS pumps were unavailable in this configuration.

C. Incorrect: This would be correct if the ADS was divisional and that the candidate misunderstood that the amber light being illuminated indicates that this logic is available for operation.

D. Incorrect: This would be correct if the ADS was divisional and would only operate thePSV-4402 and PSV-4406 due to having only “B” side essential low pressure ECCS pumps.

Technical Reference(s): SD-183-1, Rev. 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 8.02.01.02, List the signals which cause an ADS system auto initiation including setpoints and logic. Describe how they are bypassed and how they are reset. (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	

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K4.04	
	Importance Rating	3.2	

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/~~NUCLEAR STEAM SUPPLY SHUT-OFF~~ design feature(s) and/or interlocks which provide for the following: Automatic bypassing of selected isolations during specified plant conditions

Proposed Question: RO Question 41

- The plant is MODE 2
- MSIV's are OPEN
- Main Turbine is RESET
- Reactor water level is 190 inches and stable
- Reactor Pressure is 800 psig and stable
- Condenser high backpressure bypass switches are in BYPASS position
- Condenser backpressure is 10 inches Hg Abs and improving

Which of the following would cause a Group 1 isolation that would close the MSIV's?

A. Placing the MODE Switch in RUN

B. Raising reactor water level to 215 inches

C. Opening the Main Condenser Vacuum Breakers

D. Receipt of a valid Main Steam Line HI RAD annunciator

Proposed Answer: A

Explanation (Optional):

A. Correct: This PCIS logic to close the MSIV's to protect the nuclear fuel is automatically bypassed when the MODE switch is in Start & Hot STBY (MODE 2.)

B. Incorrect: A turbine trip signal will be received at a reactor water level of 211 inches and rising. This trip signal will not affect the logic necessary to keep MSIV's open if a degraded vacuum were to occur in the Main condenser. This is not for PCIS protection, this is for main condenser protection.

C. Incorrect: In this plant configuration, the Condenser high backpressure bypass switches are in BYPASS position, which will prevent MSIV to go closed of degraded main condenser vacuum.

D. Incorrect: This is a Group 1 isolation with the exception of MSIV's.

Technical Reference(s): SD-683, Rev. 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	48.00.00.02, list the signals which cause a MS isolation including purpose, setpoints, and logic. Describe how they are bypassed	(As available)

Question Source:	Bank #	19172	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K2.01	
	Importance Rating	2.8	

Knowledge of electrical power supplies to the following: SRV solenoids

Proposed Question: RO Question 42

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

A. ADS Logic is powered by its divisional 125 VDC power supply. It has no backup power

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

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

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

Proposed Answer: D

Explanation (Optional):

A. Incorrect - ADS SRV Logic B has a backup power supply.

B. Incorrect - ADS SRV Logic A has NO backup power supply

C. 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)

D. 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 de-energize 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.

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

Proposed References to be provided to applicants during examination: N

Learning Objective: 8.01.01.02, given an ADS system operating mode and various plant conditions, predict how the ADS (As available)

(SRV) system will be impacted by failures in the following support systems: d. 125 VDC buses 1D1 and 1D2

Question Source:	Bank #	46977	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2009 NRC

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

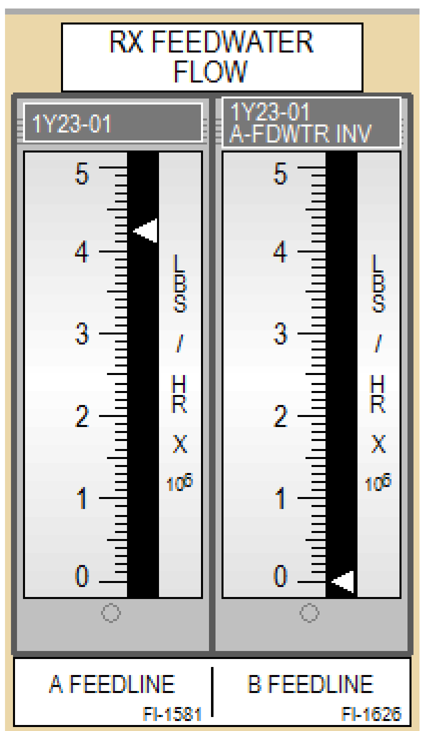
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K6.04	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor feedwater flow input

Proposed Question: RO Question 43

- The plant was operating at full reactor power

The Operator at the Controls (OATC) observes the following indication:



The Operator should place __________ to stabilize the transient.

A. HSS-4450, 1 or 3 Element Control Select to 1 ELEMENT

B. HSS-4450, 1 or 3 Element Control Select to 3 ELEMENT

C. HC-1621, B Feed REG Valve Manual/AUTO Transfer, controller in manual

D. HC-1622, Startup Feed REG Valve Manual/AUTO Transfer, controller in manual

Proposed Answer: A

Explanation (Optional):

A. Correct – Reactor water level will rise and placing HSS-4450 to 1 ELEMENT will remove the effects caused by the steam flow/feed flow mismatch created by the flow transmitter sending an incorrectly LOW flow signal.

B. Incorrect – HSS-4450, 1 or 3 Element Control Select is already in 3 element.

C. Incorrect –. Taking the Master Feed Reg Valve Controller in manual will only stop the feed regulating valves from opening further. With the Feed Regulating Valves already further open than necessary, reactor water level will continue to rise.

D. Incorrect –Taking HC-1622, Startup Feed REG Valve Manual/AUTO transfer, to manual will not stabilize the transient.

Technical Reference(s): SD-644, Rev. 16 (Attach if not previously provided)

AOP 644, Rev. 17

Proposed References to be provided to applicants during examination: N

Learning Objective: 94.56.03.02, Evaluate plant conditions and control room indications to determine the required operator actions of AOP 644. (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K4.01	
	Importance Rating	3.7	

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation

Proposed Question: RO Question 44

Which of the following annunciators, if alarming on a VALID signal, would automatically initiate Standby Gas Treatment?

A. Post Treat RR-4101 HI-HI Rad

B. Primary Containment HI/LO Pressure

C. Offgas Vent Pipe RM-4116A/B HI-HI Rad

D. Reactor Vessel HI/LO Level Recorder Alarm

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Post Treat RR-4101 HI-HI Rad does not alarm at a Group 3 setpoint.

B. Incorrect: Primary Containment HI/LO Pressure annunciator on high containment pressure alarms at 1.5 psig. The Group 3 and automatic initiation of Standby Gas Treatment is at 2 psig.

C. Correct: Offgas Vent Pipe RM-4116A/B HI-HI Rad alarms at a Group 3 setpoint. This alarm threshold is an input into the Group 3 isolation logic which when reached will automatically initiate Standby Gas Treatment.

D. Incorrect: Reactor Vessel HI/LO Level Recorder Alarm annunciator on low reactor water level is at 196 inches. The Group 3 and automatic initiation of Standby Gas Treatment is at 170 inches.

Technical Reference(s): ARP 1C03A, Rev. 58 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 7.02.01.02, list the signals which cause a SBGT System auto initiation including setpoints and logic. (As available)

Describe how they are bypassed and
how they are reset.

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

Question History: Last NRC Exam:

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

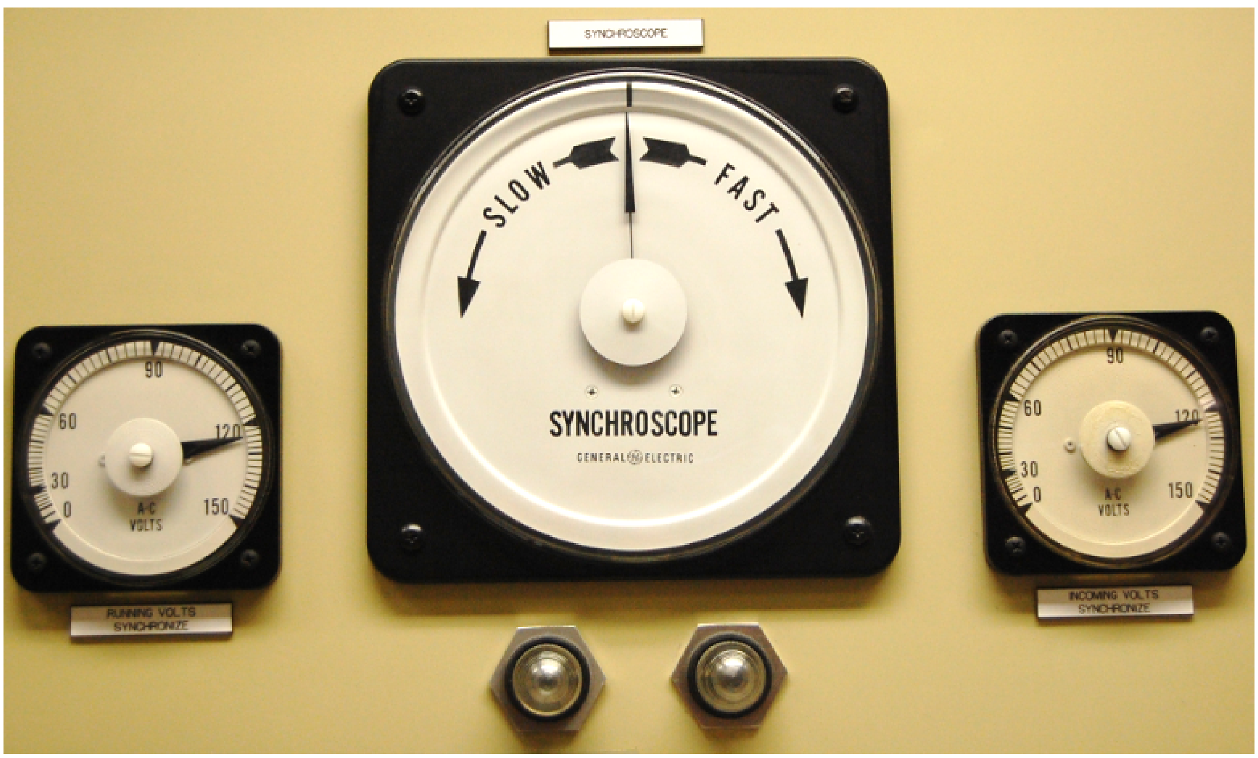
Comments: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 A4.02	
	Importance Rating	3.4	

Ability to manually operate and/or monitor in the control room: Synchroscope, including understanding of running and incoming voltages [A.C. Electrical Distribution]

Proposed Question: RO Question 45

- The Main Generator is being synchronized to the grid in accordance with OI 698, Main Generator System
- The synchroscope is rotating at 3 RPM
- The synchroscope is rotating in the slow direction



Which of the following is correct for the current status of the Main Generator?

The Operator ______.

A. has no adjustments to make and the main generator is ready to be paralleled to the grid

B. cannot close the main generator output breaker due to a failure of the synch check relays lights

C. will have to lower the running voltage and lower the speed to parallel the main generator to the grid

D. will have to raise the incoming voltage and raise the speed to parallel the main generator to the grid

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Adjustments are necessary to ensure that the main generator does not reverse power. This will require that raising machine voltage and speed (frequency) for the machine to complete a proper parallel with the grid. The candidate will select this answer if they have the misconception of the requirements to parallel the machine to the grid.

B. Incorrect: The synch check relay lights should be extinguished when the frequency between the main generator and the grid are within a set tolerance (the picture demonstrates this tolerance is met.) The candidate will select this answer if they have the misconception that the lights should be illuminated to demonstrate that the main generator and the grid are within frequency tolerances.

C. Incorrect: The demonstrated main generator voltage shows approximately 21.5 kV AC which is required to be at 22kV. This needs to be corrected to 22 kV which will raise the incoming volts snchronize indication (required to be approximately the same as running volts synchronize. The information provided on the direction of rotation of the syncroscope will require the Operator to raise speed of the machine to change the rotation to “slow in the fast direction” at an RPM of less than 1.5 RPM by the Operating Instruction. The candidate will select this answer if they have the misconception that they need to lower running voltage and if they believe that the syncroscope is rotating the correct direction.

D. Correct: The demonstrated main generator voltage shows approximately 21.5 kV AC which is required to be at 22kV. This needs to be corrected to 22 kV which will raise the incoming volts snchronize indication (required to be approximately the same as running volts synchronize. The information provided on the direction of rotation of the syncroscope will require the Operator to raise speed of the machine to change the rotation to “slow in the fast direction” at an RPM of less than 1.5 RPM by the Operating Instruction.

Technical Reference(s): OI 698, Rev. 97 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 57.00.00.02, Evaluate plant conditions and control room indications to determine if the Main Generator System is operating as expected, and identify any actions that may be (As available)

necessary to place the Main
Generator System in the correct lineup

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 K1.01	
	Importance Rating	2.8	

Knowledge of the physical connections and/or cause effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Feedwater Level Control

Proposed Question: RO Question 46

Which valve(s) can be throttled to control Reactor Water level on a loss of 1Y23, 120 VAC Uninterruptible power?

A. CV-1622, Startup Feed Regulating valve

B. CV-1570 and CV-1621, “A” and “B” Feed Regulating valves

C. MO-1592 and MO-1636, “A” and “B” Feedline Block valves

D. MO-4441 and MO-4442, “A” and “B” Feedwater Line Reactor Inlet valves

Proposed Answer: C

Explanation (Optional):

A. Incorrect: On a loss of 1Y23, 120 VAC Uninterruptible power the Startup Feed Regulating Valve controller has lost power.

B. Incorrect: On a loss of 1Y23, 120 VAC Uninterruptible power the “A” and “B” Feed Regulating Valves have a silent lockup and power is lost to the controllers.

C. Correct: The “A” and “B” Feedline Block valves are able to be throttled to control reactor water level. This action is directed from AOP 358,

D. Incorrect: The “A” and “B” Feedwater Line Reactor Inlet valves cannot be throttled.

Technical Reference(s): AOP 357, Rev. 46 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 94.10.01.02, Relate the automatic actions of a loss of 1Y23 to the immediate and follow-up actions directed by AOP 357. (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	

Comments: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

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 47

- The plant is operating at full reactor power

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

A. Both Instrument AC buses would transfer to a Regulating Transformer 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 a loss of all Uninterruptible AC power

Proposed Answer: C

Explanation (Optional):

A. Incorrect: This would be true if 1D10 and 1D20 125 VDC buses were lost. The transfer would perform as designed and the plant would remain stable.

B. Incorrect: This would be true if 1D10 and 1D20 125 VDC buses were lost. If the transfer did not perform as designed and the crew would have inserted a scram due to loss of main condenser vacuum.

C. Correct: With a loss of 1D40, 250 VDC, Uninterruptible would automatically transfer to 1Y4 and the plant would remain stable.

D. Incorrect: This would be true if the power source transfer of the static transfer switch did not occur. Operator action would be required to stabilize the plant transient.

Technical Reference(s): SD-388, Rev. 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	21.00.00.04d, Describe the purpose and operation of the following Uninterruptible AC system components: Static Switch	(As available)

Question Source:	Bank #	46983	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2009

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K3.02	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers)

Proposed Question: RO Question 48

- 125 VDC bus 1D11 is de-energized

The 1A1, 4160 VAC Nonessential Switchgear, 4KV breaker overcurrent and undervoltage protection is ___(1)___ and the 4KV breakers will ___(2)___.

	(1)	(2)
A.	lost	trip open
B.	lost	remain closed
C.	available	trip open
D.	available	remain closed

Proposed Answer: B

Explanation (Optional):

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

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

C. Incorrect – The breakers remain closed without breaker protection.

D. Incorrect – This would be true if the 125 VDC bus was not 1D11.

Technical Reference(s): AOP 302.1, Rev. 57 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 13.00.00.05 (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	

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

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 in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Proposed Question: RO Question 49

- The Division 1 125 VDC battery charger is being operated in the equalize mode

In equalize, the charger output to the battery will be a ___(1)___ voltage than when in the float mode.

The 125 VDC batteries are sized to supply emergency power for a(n) ___(2)___ hour time period.

	(1)	(2)
A.	lower	4
B.	lower	8
C.	higher	4
D.	higher	8

Proposed Answer: C

Explanation (Optional):

A. Incorrect: When the 125 VDC battery charger is selected to “Equalize” nominal equalizing voltage will raise to approximately 137.0 to 137.5 VDC. This value is greater than the nominal “Float” voltages of 133 to 136 VDC found in the Operating Instructions. The candidate will select this answer if they have the misconception of the term equalize meaning the same as battery voltage. This term is specific to electrical systems in that the meaning is to raise voltage to charge a battery.

B. Incorrect: When the 125 VDC battery charger is selected to “Equalize” nominal equalizing voltage will raise to approximately 137.0 to 137.5 VDC. This value is greater than the nominal “Float” voltages of 133 to 136 VDC found in the Operating Instructions. The candidate will select this answer if they have the misconception of the term equalize meaning the same as battery voltage. This term is specific to electrical systems in that the meaning is to raise voltage to charge a battery. The candidate will select this answer if they have the misconception of the goal of the FLEX strategies which has the in plant Operator strip the DC electrical buses to extend DC power for the coping period specified by FLEX.

C. Correct: When the 125 VDC battery charger is selected to “Equalize” nominal equalizing voltage will raise to approximately 137.0 to 137.5 VDC. This value is greater than the

nominal “Float” voltages of 133 to 136 VDC found in the Operating Instructions. The station batteries have been sized appropriately to allow full system usage for approximately 4 hours.

D. Incorrect: The station batteries have been sized appropriately to allow full system usage for approximately 4 hours. The candidate will select this answer if they have the misconception of the goal of the FLEX strategies which has the in plant Operator strip the DC electrical buses to extend DC power for the coping period specified by FLEX.

Technical Reference(s): SD-375, Rev. 9 (Attach if not previously provided)

Battry-C173-01<sec B>, Rev. 38

Proposed References to be provided to applicants during examination: N

Learning Objective: 13.00.00.05, evaluate plant conditions and Control Room indications to determine if the 125 VDC power system is functioning as expected and identify any actions that may be necessary to place the system in the correct condition (As available)

Question Source: Bank # 46984

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K5.06	
	Importance Rating	3.4	

Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Load sequencing

Proposed Question: RO Question 50

Which of the following describes the load sequencing of the Standby Diesel Generators (SBDG)s following a complete instantaneous loss of offsite power?

(Assume normal full power operation and SBDGs in standby readiness as initial conditions.)

SBDG picks up the 1A3 and 1A4 buses _____(1)_____ after the loss of power.

Emergency Service Water Pump 1P-99A(B) starts _____(2)_____ after the SBDG picks up the 1A3 and 1A4 buses.

	(1)	(2)
A.	≤10 seconds	immediately
B.	≤10 seconds	5 seconds
C.	≤18.5 seconds	immediately
D.	≤18.5 seconds	5 seconds

Proposed Answer: A

Explanation (Optional):

A. Correct - The time frame assumed in the UFSAR and the Technical Specifications for the SBDGs to start and reenergize the essential busses is ≤10 seconds). The ESW pump starts immediately when power is restored to the bus.

B. Incorrect - The ESW pump start immediately when power is restored to the bus.

C. Incorrect - The ESW pump starts immediately when power is restored to the bus. The 18.5 seconds used as a distracter is based on the 8 to 8.5 second delay to start the SBDG during a degraded voltage situation. In this question the stem specifies a complete instantaneous loss of offsite power, therefore the time delay does NOT apply.

D. Incorrect - The ESW pump starts immediately when power is restored to the bus. The 18.5 seconds used as a distracter is based on the 8 to 8.5 second delay to start the SBDG during a degraded voltage situation. In this question the stem specifies a complete instantaneous loss of offsite power, therefore the time delay does NOT apply.

Technical Reference(s): SD-324, Rev.16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	19.01.01.07, Explain the SBDG starting logic for a manual or automatic start signal including setpoints.	(As available)

Question Source:	Bank #	46985	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: PDA 2013

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments: 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.

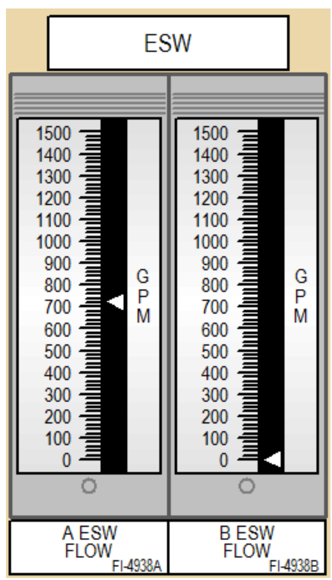
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A3.06	
	Importance Rating	3.1	

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Cooling water system operation

Proposed Question: RO Question 51

- The plant is operating at full reactor power
- “A” Emergency Service Water (ESW) was started
- “A” Standby Diesel Generator (SBDG) is running at full load for a surveillance

The Operator observes the following ESW flows:



The “A” SBDG will _______.

A. trip on exhaust high temperature

B. trip on high jacket coolant temperature

C. not trip because ESW does not cool the “A” SBDG

D. not trip as this is expected ESW flow for system operation

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This is not a diesel trip however the Operator will have to take manual action of high exhaust temperature.

B. Correct: The diesel, without a LOOP or LOCA signal (which would override this trip), will cause the diesel to automatically shutdown with 2 of 3 temperature indications at 1950F, 2000F, and 2050F.

C. Incorrect: Both SBDG's are cooled by ESW. This could be mistaken since there are multiple diesel engines that have their own independent cooling systems.

D. Incorrect: The indicated flow is expected for the initial start of ESW. When a SBDG gets a start signal, flow will to approximately 1100 gpm.

Technical Reference(s): SD-324, Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 33.00.00.04, Evaluate plant conditions and control room indications to determine if the ESW System is operating as expected, and identify any actions that may be necessary to place the ESW System in the correct lineup (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K2.02	
	Importance Rating	3.0	

Knowledge of electrical power supplies to the following: Emergency air compressor

Proposed Question: RO Question 52

Which of the following electrical buses, if lost, would de-energize 1K003, CB/SBGT Instrument Air Compressor?

A. 1B32, Control Building 480 VAC Motor Control Center

B. 1B33, Turbine Building 480 VAC Motor Control Center

C. 1B42, Control Building 480 VAC Motor Control Center

D. 1BR91, INST Air Building 480 VAC Motor Control Center

Proposed Answer: A

Explanation (Optional):

A. Correct: This is a power supply to 1K003, CB/SBGT Instrument Air Compressor

B. Incorrect: This is a power supply to 1K001, Backup Air Compressor

C. Incorrect: This is a power supply to 1K004, CB/SBGT Instrument Air Compressor

D. This is a power supply to 1K090A, Instrument Air Compressor

Technical Reference(s): AOP 301, Rev. 71 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 36.00.00.05, Evaluate plant conditions and control room indications to determine if the Instrument and Service Air System is operating as expected, and identify any actions that may be necessary to place the Instrument and Service Air System in the correct lineup. (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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 K1.02	
	Importance Rating	3.2	

Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Loads cooled by CCWS

Proposed Question: RO Question 53

- The plant is operating at full reactor power
- A complete loss of offsite power occurs

With the following plant conditions:

- 1K1, Backup Instrument and Service Air Compressor, is aligned to 1B45
- Breaker 1B302, CB 480VAC Motor Control Center 1B33, has tripped

The 1K1 air compressor has ___(1)___ and 1K3 and 1K4 air compressors have ___(2)___.

A. (1) cooling water flow
(2) cooling water flow

B. (1) no cooling water flow
(2) cooling water flow

C. (1) cooling water flow
(2) no cooling water flow

D. (1) no cooling water flow
(2) no cooling water flow

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The running well water pumps will lose power when 1B33 loses power due to the LOOP and 1B33 breaker trip.

B. Correct: The running well water pumps will lose power when 1B33 loses power due to the LOOP and 1B33 breaker trip. They will not restart without Operator action. 1K-3 and 1K-4 will be cooled with ESW which will be available when the SBDG's re-energize their respective busses.

C. Incorrect: 1K-3 and 1K-4 will be cooled with ESW which will be available when the SBDG's re-energize their respective busses.

D. Incorrect: The running well water pumps will lose power when 1B33 loses power due to the LOOP and 1B33 breaker trip. They will not restart without Operator action. 1K-3 and 1K-4 will be cooled with ESW which will be available when the SBDG's re-energize their respective busses.

Technical Reference(s):	SD-408, Rev. 12	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	36.00.00.05, Evaluate plant conditions and control room indications to determine if the Instrument and Service Air System is operating as expected, and identify any actions that may be necessary to place the Instrument and Service Air System in the correct lineup.	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 K6.03	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Rod position indication: P-Spec(Not-BWR6)

Proposed Question: RO Question 54

- The plant is operating at full reactor power
- A control rod needs to be inserted one notch

When the Operator selects the control rod, the control rod position switch fails OPEN.

The Operator will ________________ to insert the control rod.

A. install a RPIS jumper at 1C27

B. bypass the rod at the RWM-CC

C. have no additional actions to perform

D. substitute the rod position at the RWM-OD

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Operations will install a jumper during refueling to allow the one rod permissive function to work for a control rod that has been removed from the core.

B. Incorrect: This action is required if the Operating crew is to remove a control rod from being enforced in a sequence. This will prevent the RWM from enforcing a RWM Rod Block.

C. Incorrect: Although the RWM is not required to be operable at this power level (>10%) the RWM is continuously enforcing at DAEC and operator action is required to bypass the control rod position.

D. Correct: The Operator will have to substitute the rod position using the RWM-OD to allow the RWM to move the control rod to another position that has an functional reed switch.

Technical Reference(s): OI 878.8, Rev. 27 (Attach if not previously provided)

SD-878-8, Rev. 9 ARP 1C05B(D-6), Rev. 105

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.02.01.04, evaluate plant conditions and control room indications to determine the required operator actions	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 K1.28	
	Importance Rating	3.9	

Knowledge of the physical connections and/or cause effect relationships between RECIRCULATION SYSTEM and the following: End-of-cycle recirculation pump trip circuitry: Plant-Specific

Proposed Question: RO Question 55

- The plant is operating at 67% reactor power
- The crew is performing a surveillance on EOC-RPT logic

The current status of EOC-RPT Logic:

- C71A-S12A, "A" RPT Mode Select Switch, on 1C15 is in BYPASS
- C71A-S12B, "B" RPT Mode Select Switch, on 1C17 is in NORMAL

A complete loss of EHC system pressure occurs.

What is the response of the Recirculation System?

	"A" Recirc pump	"B" Recirc pump
A.	trip	trip
B.	trip	running
C.	running	trip
D.	running	running

Proposed Answer: A

Explanation (Optional):

A. Correct: EOC-RPT setpoint will be reached with a loss of EHC pressure. When EHC pressure lowers to 900 psig RETS, the EOC-RPT logic will send a trip signal to both recirculating pumps by tripping open breakers 1A501 and 1A502.

B. Incorrect: The "B" recirculating pump will receive a trip signal.

C. Incorrect: The "A" recirculating pump will receive a trip signal.

D. Incorrect: Both "A" and "B" recirculating pumps will receive a trip signal.

Technical Reference(s): SD-264, Rev. 13: Figure #12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	12.00.00.03, Describe the operation of the following principle recirc system components: c. RPT breakers	(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	6
	55.43	

Comments: Design, components, and function of reactivity control mechanisms and instrumentation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002 K4.03	
	Importance Rating	3.0	

Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Signal failure detection: Plant-Specific

Proposed Question: RO Question 56

The plant is operating at full reactor power.

- Annunciator 1C04A(C-5), 'A' RECIRC MG Scoop Tube Lock, is received
- Total Core Flow, on 1C05, has RAISED slightly and stabilized

With these conditions, which one of the following caused the scoop tube lock?

A. High Speed Demand vs. Positioner Feedback

B. Loss of 1Y23, 120 VAC Uninterruptible Power

C. Loss of 1Y11, 120 VAC Instrument Control Power

D. A Recirc MG Fluid Drive Oil Temperature at 210°F

Proposed Answer: A

Explanation (Optional):

A. Correct: High Speed Demand vs. Positioner Feedback will generate a rise in total core flow and the scoop tube will lock at 4.6% scaled 20-100% or 5.75% scaled 0-100% deviation to prevent a runaway recirc pump.

B. Incorrect: A loss of 1Y23 will cause both recirc pumps to runback to 20% (lowering total core flow) and it will not cause a recirc scoop tube lockout.

C. Incorrect: A loss of 1Y11 will cause a scoop tube lock without a change in indicated Total Core Flow.

D. Incorrect: Recirc MG Fluid Drive Oil Temperature at 210°F will cause a scoop tube lock and the MG Set Drive Motor breaker to trip.

Technical Reference(s): SD-264, Rev. 13 (Attach if not previously provided)

ARP 1C04A, Rev. 62

Proposed References to be provided to applicants during examination: N

Learning Objective:	10.00.00.02		(As available)
Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	2015 PDA LOIT Audit Exam	Last NRC Exam:	

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

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

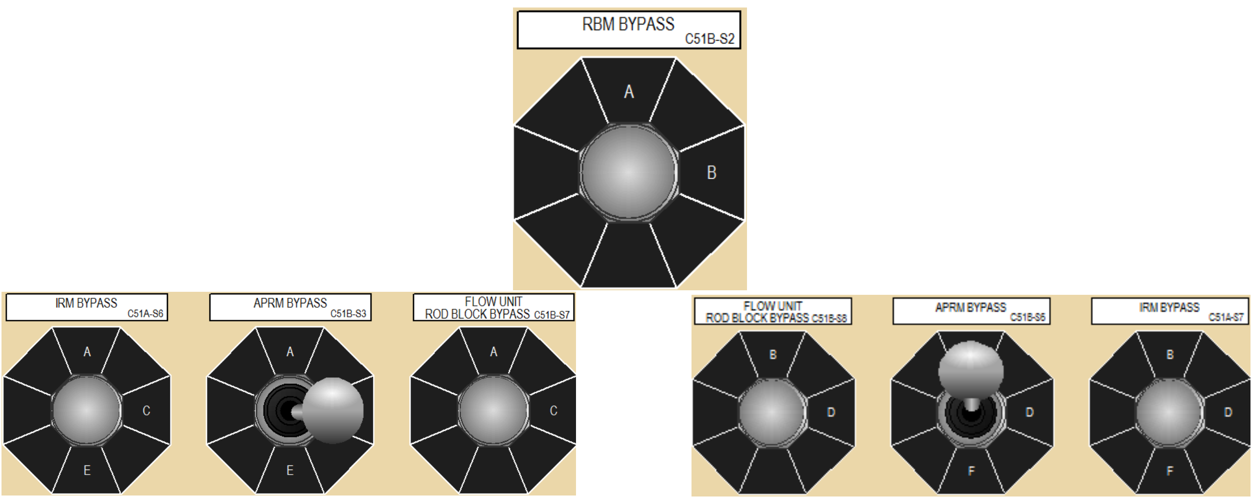
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002 K5.01	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to ROD BLOCK MONITOR SYSTEM: Trip reference selection: Plant-Specific

Proposed Question: RO Question 57

- The plant is operating at full reactor power

The following is the switch lineup for Rod Block Monitor (RBM):



- A center control rod has been selected
- “D” ARPM output failed to 20% reactor power

Which one identifies the ability of RBM Channel “A” and “B” to generate a RBM ROD BLOCK?

	RBM Channel “A”	RBM Channel “B”
A.	can	can
B.	can	cannot
C.	cannot	can
D.	cannot	cannot

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Would be true if a non-referenced APRM failed downscale.

B. Correct: APRM “D” is the reference APRM for RBM Channel “B”. If APRM “D” fails downscale, it would provide a signal to the RBM “B” that is below 30% power. This would result in RBM “B” being automatically bypassed, so no blocks will be generated. RBM “A” is not affected and can still generate blocks.

C. Incorrect: Would be true if APRM “A” failed downscale.

D. Incorrect: Would be true if a peripheral rod were selected.

Technical Reference(s): SD-878-5, Rev. 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 82.01.01.02, Given a Rod Block Monitor System operating mode and various plant conditions, predict how the Rod Block Monitor System will be impacted by failures in the following support system: B. APRM (As available)

Question Source: Bank #

Modified Bank # 48749 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments: 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.

Modified to evaluate “B” RBM using “D” APRM as the failing reference signal. This took answer choice “C” as being correct to answer choice “B” as now correct.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	223001 K4.01	
	Importance Rating	3.7	

Knowledge of PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES design feature(s) and/or interlocks which provide for the following: Allows for absorption of the energy released during a LOCA

Proposed Question: RO Question 58

The plant was operating at full reactor power when a design basis Loss of Coolant Accident (LOCA) occurs coincident with a Loss of Offsite Power (LOOP.)

Which one of the following primary containment design features directly facilitates steam condensation to protect containment in this condition?

A. Drywell Air Coolers

B. SRV T-quencher spargers

C. Sufficient Torus Water Level

D. Torus to Drywell Vacuum Breakers

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The drywell air coolers will not operate correctly with a loss of offsite power.

B. Incorrect: This is for steam dissipation from an SRV only.

C. Correct: The downcomers direct steam under the Torus water line to allow steam condensation to occur.

D. Incorrect: The Torus to Drywell Vacuum Breakers are required to release non-condensable gases back to the Drywell air space to prevent containment damage.

Technical Reference(s): SD-959, Rev. 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 42.00.00.04, Describe the functions of the following Primary Containment and Containment Atmosphere (As available)

Monitoring and Control System
components: b. Suppression
chamber/Pool and internals

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

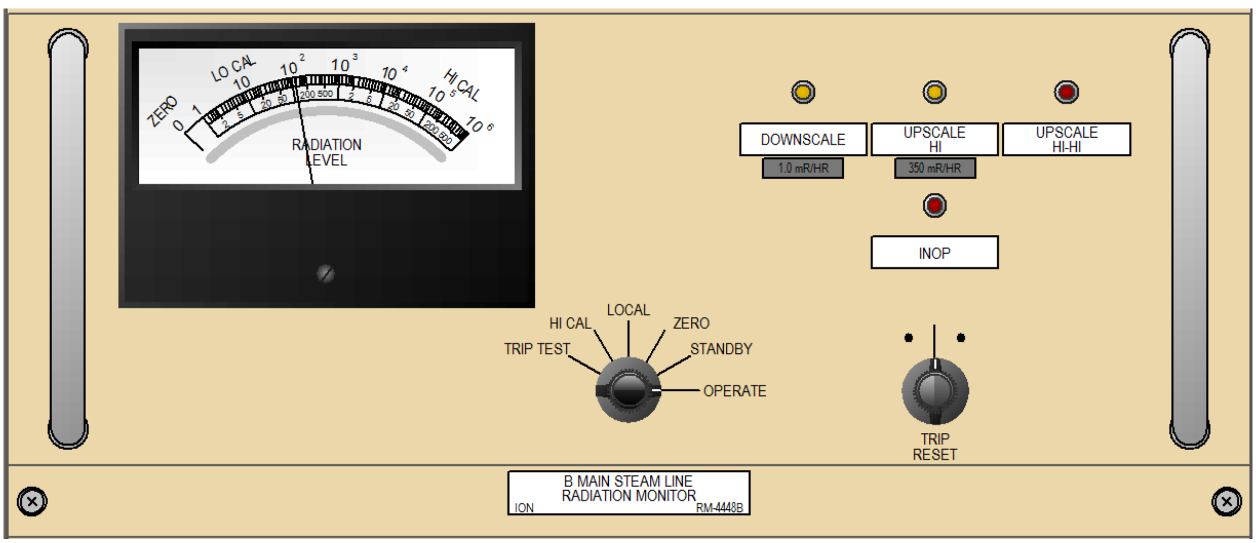
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	239001 A1.05	
	Importance Rating	3.6	

Ability to predict and/or monitor changes in parameters associated with operating the MAIN AND REHEAT STEAM SYSTEM controls including: Main Steam Line Radiation Monitors

Proposed Question: RO Question 59

- The plant is operating at 50% reactor power

The following is observed at 1C36:



NOTE: Analyze each condition separately.

If the “B” Main Steam Line is isolated at this reactor power level, RM-4448B will indicate ___(1)___ than the above indicated radiation level.

When the plant is brought to full reactor power, RM-4448B will indicate ___(2)___ than the above indicated radiation level.

	(1)	(2)
A.	less	less
B.	less	greater
C.	greater	less

D. greater greater

Proposed Answer: B

Explanation (Optional):

A. Incorrect: RM-4448B indicated radiation will indicate lower when reactor power is lowered.

B. Correct: If the “B” Main Steam Line is isolated, N-16 radiation levels will lower. As reactor power and subsequent steam line flows, N-16 concentration will raise in the steam tunnel and at the radiation monitor.

C. Incorrect: With the “B” Main Steam Line isolated, more steam will be conducted through the other 3 available steam lines. With this raised steam flow, it could be misapplied that radiation levels from N-16 would raise indicated radiation levels in the steam tunnel. RM-4448B indicated radiation will indicate lower when reactor power is lowered.

D. Incorrect: With the “B” Main Steam Line isolated, more steam will be conducted through the other 3 available steam lines. With this raised steam flow, it could be misapplied that radiation levels from N-16 would raise indicated radiation levels in the steam tunnel.

Technical Reference(s): SD-879.1, Rev. 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 85.00.00.05, Evaluate plant conditions and control room indications to determine if the PRM system is operating as expected and identify any actions that may be necessary to place the PRM system or the plant in the correct conditions. (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

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000 A3.04	
	Importance Rating	3.0	

Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: System Flow

Proposed Question: RO Question 60

The plant is operating at 50% reactor power for a load line adjustment.

- “A” and “B” Condensate pumps are operating
- “A” and “B” Reactor Feed pumps (RFP) are operating

The discharge pipe of the “B” Condensate pump ruptures and causes the following:

- “B Condensate pump has tripped
- “A” RFP suction pressure lowered to 180 psig and stabilized

The Operator at the Controls should ____.

A. insert a manual reactor scram

B. stabilize Reactor Water Level

C. start “A” Reactor Feed pump

D. start “B” Condensate pump

Proposed Answer: A

Explanation (Optional):

A. Correct: With the replacement of the Reactor Feed Pumps the low suction pressure trips have been re-installed. The “B” RFP tripped due to condensate/feedpump interlock. The “A” RFP will trip at the low suction pressure of 225 psig for 5 seconds. This will give a total loss of feedwater and a manual reactor scram is required per AOP 644.

B. Incorrect: This would be correct if the “A” RFP did not trip on low suction pressure.

C. Incorrect: With the low suction pressure for the “A” RFP, the pump would not stay operating.

D. Incorrect: With the “B” Condensate pump tripped, it will not be able to be started from the control room until the fault has been investigated.

Technical Reference(s): AOP 644, Rev. 17 (Attach if not previously provided)

SD-644, Rev. 16

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.56.03.02, Evaluate plant conditions and control room indications to determine the required operator actions of AOP 644.	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	268000 G2.4.4	
	Importance Rating	4.5	

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures: Radwaste

Proposed Question: RO Question 61

The plant is operating at full reactor power when the following annunciator is received during a resin transfer:

- 1C35A (B-3), LLRPSF KAMAN 12 HI RAD or Monitor Trouble

Effluent Monitoring System (EMS) is indicating that a release is in progress that meets the ALERT level threshold from the Low Level Rad Waste Shipping and Storage Facility.

Which, if any, procedure will be entered and what will direct securing the release in progress?

A. No EOP entry is required the release will be secured using the Annunciator Response Procedure actions

B. EOP 4, Radioactivity Release Control, the release in progress will be secured using Annunciator Response Procedure actions

C. Emergency Management Guideline will direct isolating all systems discharging into the area except needed for EOP and/or damage control

D. EOP 3, Secondary Containment Control will direct isolating all systems discharging into the area except needed for EOP and/or damage control

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The threshold for EOP 4 has been reached with a release concentration that of an ALERT.

B. Correct: The threshold for EOP 4 has been reached with a release concentration that of an ALERT. The annunciator response will direct that the ventilation would need to be secured to terminate the release.

C. Incorrect: Emergency Management Guides would be utilized since an EAL threshold has been exceeded, however the direction from the EMG is to enter the applicable EOP board and isolate the system per the EOP boards. The isolation of the release is directed from the ARP.

D. Incorrect: EOP 3 would not be used as this release is occurring outside secondary containment. The isolation of the release is directed from the ARP.

Technical Reference(s):	ARP 1C35A, Rev. 46	(Attach if not previously provided)
Bases-EOP 4, Rev. 10		

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.23, Determine which EOP entry conditions exists	(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	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	272000 A4.06	
	Importance Rating	2.9	

Ability to manually operate and/or monitor in the control room: Manually trip process radiation monitor logic [Radiation Monitoring System]

Proposed Question: RO Question 62

The “A” Standby Gas Treatment (SBGT) System is going to be used to verify secondary containment integrity.

NOTE: Components with their associated Noun Name

PB-7606A	“A” Group 3 Initiation push button
PB-5831A	SBGT Train “A” Test push button
L/R-5830A	Lockout Relay

To perform this verification it will require the Operator to _______.

A. depress PB-5831A which will start A SBGT and insert only a secondary containment isolation

B. depress PB-5831A which will start A SBGT and then manually trip L/R-5830A to insert only a secondary containment isolation

C. depress PB-7606A which will start A SBGT and insert a primary and secondary containment isolation

D. depress PB-7606A which will insert a primary and secondary containment isolation and then depress PB-5831A which will start A SBGT

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Depressing PB-5831A will start A SBGT however it will not insert the needed secondary containment isolation needed for the verification.

B. Incorrect: To initiate the verification, does not require starting the A SBGT train using PB-5831A and manually tripping L/R-5830A. L/R-5830A is an input to Primary Containment Isolation not Secondary Containment.

C. Correct: Depressing PB-7606A will automatically start A SBGT and inserting a primary and secondary Group 3 isolation. This is accomplished by tripping the RB Vent Shaft Process Radiation Monitor logic to simulate 8 mr/hr.

D. Incorrect: To initiate the verification, does not require both starting the A SBGT train using PB-5831A and PB-7606A.

Technical Reference(s): SD-170, Rev. 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 7.04.01.03, describe how the SBGT System responds to a Group 3 PCIS isolation signal (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000 K2.02	
	Importance Rating	2.9	

Knowledge of electrical power supplies to the following: Pumps [Fire Protection System]

Proposed Question: RO Question 63

- A fire results in a loss of 1B1, Turbine Building 480 VAC Load Center

What is the availability of the Fire Protection system pumps?

	Diesel	Electric	Jockey
A.	available	available	-
B.	-	available	available
C.	-	-	available
D.	available	-	-

Proposed Answer: D

Explanation (Optional):

A. Incorrect: This would be correct if the Electric fire pump was powered from a different 480V power source.

B. Incorrect: This would be correct if the Electric and Jockey fire pump was powered from a different 480V power source.

C. Incorrect: This would be correct if the Jockey fire pump was powered from a different 480V power source.

D. Correct: The Jockey Fire Pump 1P-47 receives power from MCC 1B13 (breaker 1B1311) and the Electric Fire pump 1P-48 receives power from MCC 1B1 (breaker 1B0106.) The Diesel Fire Pump 1P-49 battery charger and control panel 1C116 receives power from 1L50 circuit #7.

Technical Reference(s): SD-513, Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 34.05.01.06 (As available)

Question Source:	Bank #	19106	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

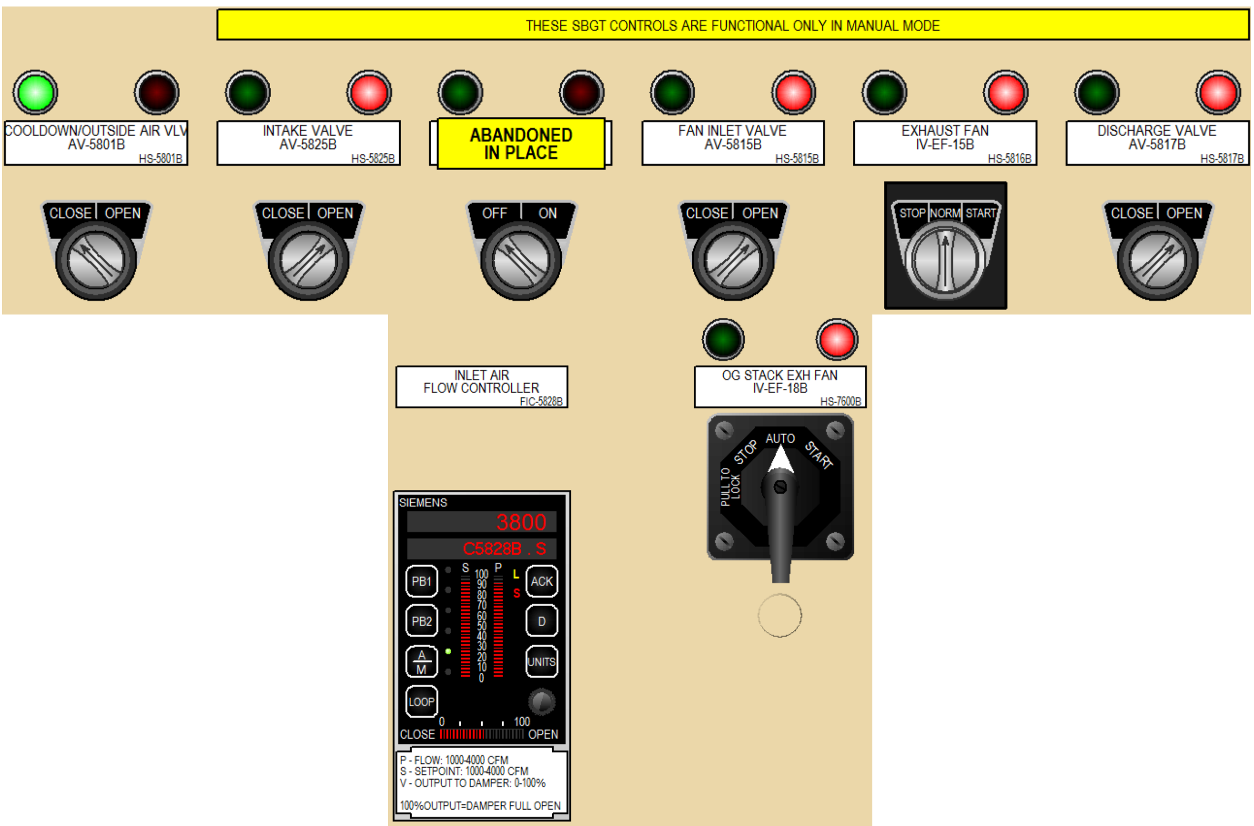
Comments: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001 K3.01	
	Importance Rating	4.0	

Knowledge of the effect that a loss or malfunction of the SECONDARY CONTAINMENT will have on following: Off-site radioactive release rates

Proposed Question: RO Question 64

- The reactor is shutdown
- Fuel damage has occurred
- An unisolable primary leak is occurring from HPCI steam line
- The “A” Standby Gas Treatment System is unavailable
- Reactor Building D/P is -0.20 inches of water



What action should the Operator take to reduce Off-site radioactive release rates?

A. Secure 1V-EF-18B, OG Stack Exh Fan

B. Open AV-7604U, Fuel Pool to SBGT Inlet

C. Open AV-5801B, Cooldown/Outside Air Valve

D. Raise the setpoint on FIC-5828B, Inlet Air Flow Controller

Proposed Answer: D

Explanation (Optional):

A. Incorrect: This would be true if the OG Stack Exhaust Fan created additional backpressure on the SBGT.

B. Incorrect: This would improve air flow through the SBGT. This action is authorized if SBGT air flow is <2400 SCFM. From the picture the process flow is approximately 3700 SCFM.

C. Incorrect: This would improve air flow through the SBGT, but it would not improve the differential pressure of the Reactor Building.

D. Correct: The controller is indicating that the SBGT train is operating at 3800 SCFM, which was the old setpoint of the SBGT train. The controller and Operating Instruction (OI) 170 permits an Operator to raise SBGT flow to 4000 SCFM.

Technical Reference(s): OI 170, Rev. 65 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 7.02.01.03, evaluate plant conditions and control room indications to determine if the SBGT System is operating as expected, and identify any actions that may be necessary to place the SBGT System in the correct lineup. (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

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290002 A2.05	
	Importance Rating	3.7	

Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding thermal limits

Proposed Question: RO Question 65

A failure of both recirculation pumps has made it necessary to establish natural circulation cooldown flow. Engineering has informed the crew that recirculation loop temperatures, normally used for calculation of the cool down rate, are unreliable.

Based on the following reactor vessel pressure recordings, select the appropriate action.

TIME	PRESSURE (psig)
1300	940
1315	770
1330	660
1345	450
1400	340

A. Continue reactor vessel cool down

B. Restore cooldown rate to within limits in 30 minutes

C. Restore cooldown rate to within limits in 45 minutes

D. Restore cooldown rate to within limits in 1 hour

Proposed Answer: B

Explanation (Optional):

A. Incorrect – See justification for correct answer.

B. Correct - Over one hour the pressure drops from 940 psig to 340 psig. This correlates to a temperature drop from 539F to 431F which is a cooldown rate of 108F/hr. TS 3.4.9 required action A.1 is to restore cooldown rate to within limits in 30 minutes and Determine RCS is acceptable for continued operation within 72 hrs. This would be a ≤1 hr TS required action which is required knowledge for ROs & SROs.

C. Incorrect - See justification for correct answer.

D. Incorrect - See justification for correct answer.

Technical Reference(s): AOP 915 Rev. 57 (Attach if not previously provided)

TS 3.4.9 Rev. 255 TRM Appendix A, Rev. 0

Proposed References to be provided to applicants during examination: AOP 915 Rev. 7

Learning Objective: 94.29.03.04 (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 X

55.43

Comments: 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.2	
	Importance Rating	4.1	

Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: RO Question 66

The plant was operating at full reactor power when the crew inserted a manual reactor scram due to a loss of condensate and feed.

- Reactor water level lowered to 100 inches and is rising

The Balance of Plant (BOP) Operator noted that after HPCI started, HPCI room temperature rose rapidly to 220°F and stabilized.

- HPCI continued to operate until it automatically tripped on reactor water high level
- In-plant Operator reported that steam is visibly escaping the HPCI room door
- HPCI Steam Line Isolation valves remain OPEN at this time

Reactor water level is now 180 inches and lowering. The condensate and feed system has been restored.

What actions are required per OP-AA-100-1000, Conduct of Operations?

The BOP Operator shall inform the ______.

A. CRS/OSM and CLOSE the MSIVs to control HPCI room temperature

B. CRS/OSM and attempt to manually ISOLATE the HPCI Steam Line since it has a valid isolation signal

C. 1C05 Operator to monitor level. Do NOT ISOLATE HPCI until HPCI room temperature reaches 300°F

D. CRS/OSM and perform SEP 307 “Rapid Depressurization with Bypass Valves”, to rapidly lower RPV pressure and slow down the leak

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Closing the MSIVs does not isolate the HPCI steam line and therefore will not help control room temperature.

B. Correct – The Nuclear Station Operating Engineer (NSOE) reports to the Operations Shift Manager/Control Room Supervisor. The NSOE is responsible for all Control Room

operations including: Initiating a reactor scram or ESF actuation if automatic trip setpoints have been exceeded and the required automatic actions did not occur.

C. Incorrect – With Condensate and Feed restored, HPCI is no longer required for level operation and isolation of the HPCI steam line should be pursued once the temperature exceeded 175F.

D. Incorrect – In order to anticipate ED, one area has to be above MAX SAFE and another area has to be above MAX NORMAL and approaching MAX SAFE.

Technical Reference(s): ACP 1410.1 Rev. 105 (Attach if not previously provided)

OP-AA-100-1000, Rev. 20

Proposed References to be provided to applicants during examination: N

Learning Objective: 96.06.02.01, (As available)

Question Source:	Bank #	46350	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: PDA 17-1 Normal Ops A RO Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	X
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.28	
	Importance Rating	4.1	

Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question 67

In automatic control, HPCI will maintain a constant ___(1)___ by controlling ___(2)___ over the range of system operating pressures.

	(1)	(2)
A.	turbine speed	system flow
B.	system flow	turbine speed
C.	system flow	pump discharge pressure
D.	pump discharge pressure	turbine speed

Proposed Answer: B

Explanation (Optional):

A. Incorrect - common misconception and opposite of the actual function.

B. Correct - both controllers adjust pump speed via the steam throttle valve to maintain the pump flow constant at the prescribed setpoint.

C. Incorrect – pump discharge pressure is not an input to the controller and will vary as RPV pressure changes.

D. Incorrect - pump discharge pressure is maintained constant and will vary as RPV

Technical Reference(s): SD-152 Rev. 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 5.00.00.02, Describe the operation of the following principle HPCI System components: a. HPCI Turbine (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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.38	
	Importance Rating	3.7	

Knowledge of the station's requirements for verbal communications when implementing procedures.

Proposed Question: RO Question 68

Mechanical Maintenance is performing a procedure that requires communication with the Control Room Operators from the NW Corner Room.

- The plant is in MODE 5 during a refueling outage
- The only available means of communication is with low watt radios
- Plant specific parameters will be communicated

Which of the following is correct in accordance with site procedures?

A. Radios SHALL NOT be used in this specific area due to sensitive instrumentation

B. For this activity Three-Part communication is NOT required to be used by Maintenance personnel

C. Three-Part communication SHALL be used by Maintenance AND Operations personnel

D. Radios SHALL NOT be used in any area of the Turbine Building due to sensitive instrumentation

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Only during plant operation are hand-held radio transmitters prohibited in the Reactor Building.
B. Incorrect - Three part communication are required by maintenance when communicating with operations.
C. Correct - Operational communications over the telephone and radio SHALL employ the same practices as face-to-face communications.
D. Incorrect - Only during plant operation are hand-held radio transmitters prohibited in the Reactor Building.

Technical Reference(s): OP-AA-100-1000 Rev. 20 (Attach if not previously provided)

ACP 1406.10 Rev. 26 PI-AA-103-1000, Rev. 8

Proposed References to be provided to applicants during examination:	N	
Learning Objective:	96.07.01.15, Explain the expectations for the Control Room Watchstanders	(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	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.25	
	Importance Rating	3.2	

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question: RO Question 69

- The plant is at 50% reactor power
- The “A” Recirculation Pump is secured

Which one of the following is the lowest MCPR can reach and NOT violate the Technical Specification MCPR Safety Limit when operating in this condition?

MCPR at….

A. 1.08

B. 1.10

C. 1.11

D. 1.12

Proposed Answer: C

Explanation (Optional):

A. Incorrect – With the reactor steam dome pressure ≥686 psig and core flow ≥10% rated core flow: MCPR shall be ≥1.08 for two recirculation loop operation or ≥1.11 for single recirculation loop operation.

B. Incorrect –This would be true before the most recent change to TS 2.0 Safety Limit for two recirculation loop operation

C. Correct - With the reactor steam dome pressure ≥686 psig and core flow ≥10% rated core flow: MCPR shall be ≥1.08 for two recirculation loop operation or ≥1.11 for single recirculation loop operation.

D. Incorrect - This would be true before the most recent change to TS 2.0 Safety Limit.

Technical Reference(s): Tech Specs 2.0 Rev.244 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	1.03.15, Explain the purpose of and application of Safety Limits as they apply in the Technical Specifications	(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	X
	55.43	

Comments: 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.

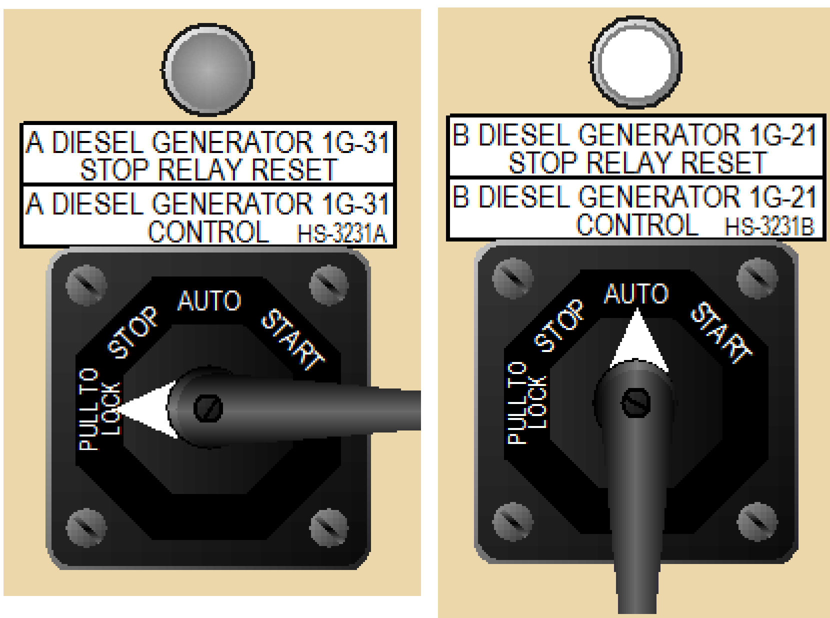
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.37	
	Importance Rating	3.6	

Ability to determine operability and/or availability of safety related equipment.

Proposed Question: RO Question 70

- The plant is operating at full reactor power

The following is the lineup of the Standby Diesel Generator (SBDG) handswitches at 1C08:



- The crew inserts a manual reactor scram
- Drywell pressure rose to 2.5 psig and stabilized

Which, if any, SBDG will start?

A. “A” SBDG

B. “B” SBDG

C. Both SBDG

D. Neither SBDG

Proposed Answer: B

Explanation (Optional):

A. Incorrect: With the handswitch at 1C08 in Pull-to-Lock, a LOCA signal will not start. This candidate would select this if they had the misconception of the white light indicating that the stop relays are reset.

B. Correct: The “B” SBDG will start when Drywell pressure rises above 2 psig.

C. Incorrect: This would be true if the candidate has the misconception that a LOCA signal will override the Pull-to-Lock handswitch position on 1C08.

D. Incorrect: This would be true if the candidate had the misconception that an autostart signal has not be reached.

Technical Reference(s): SD-324 Rev. 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 19.00.00.03, Evaluate plant conditions and control room indications to determine if the SBDG is operating as expected, and identify any actions that may be necessary to place the SBDG in the correct lineup (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: None

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.3.4	
	Importance Rating	3.2	

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: RO Question 71

10CFR20 limits the radiation exposure (TEDE) to a qualified radiation worker to ___ mrem per year.

ACP 1411.17, Occupational Dose Limits and Upgrades, limits the radiation dose (TEDE) to a qualified radiation worker to _____ mrem per year without special authorization.

	10CFR20 Limit	ACP 1411.17 Limit
A.	3000	1000
B.	3000	500
C.	5000	1000
D.	5000	500

Proposed Answer: C

Explanation (Optional):

A. Incorrect – TEDE limit is 5R. Plausible choices for lack of knowledge on radiation limits and/or approval allowances

B. Incorrect - TEDE limit is 5R exceeding 1R requires approval by Radiation Protection Manager. Plausible choices for lack of knowledge on radiation limits and/or approval allowances

C. Correct - TEDE limit is 5R exceeding 1R requires approval by Radiation Protection Manager.

D. Incorrect - 1R requires approval by Radiation Protection Manager. Plausible choices for lack of knowledge on radiation limits and/or approval allowances

Technical Reference(s): 10CFR20 (Attach if not previously provided)

ACP 1411.17 Rev. 25

Proposed References to be provided to applicants during examination: N

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	X
	55.43	

Comments: Radiological safety principles and procedures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question 72

- Annunciator 1C03A (D-9) RBCCW RM-4820 HI RAD is received
- RBCCW RAD MONITOR RIS-4820 at 1C10 verifies the condition
- RBCCW Surge Tank Level is 1 inch higher than the last recorded indication and rising

What component should be isolated to help prevent further degradation of the RBCCW system?

A. RBCCW Heat Exchangers

B. Non-Regenerative Heat Exchangers

C. Recirculation Motor Winding Heat Exchanger

D. Reactor Building Equipment Drain Sump Heat Exchanger

Proposed Answer: B

Explanation (Optional):

A. Incorrect – RBCCW Heat Exchangers are cooled by General Service Water system and would not contribute to an RBCCW HI RAD alarm.

B. Correct – As an intermediary between the primary bulk coolant and RBCCW and also at higher pressure, a leak from this component would cause both a level and radiation increase.

C. Incorrect – RBCCW cools the Reactor recirculation pump heat exchangers, which consist of seal jacket coolers and the upper and lower motor bearing oil coolers.

D. Incorrect – RBCCW is at a higher pressure thus this would not contribute to a rise in RBCCW Surge Tank level.

Technical Reference(s): SD-414 Rev. 9 (Attach if not previously provided)

ARP 1C03A Rev. 58 1C06B Rev. 64

Proposed References to be provided to applicants during examination: N

Learning Objective: 29.00.00.04 (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	12
	55.43	

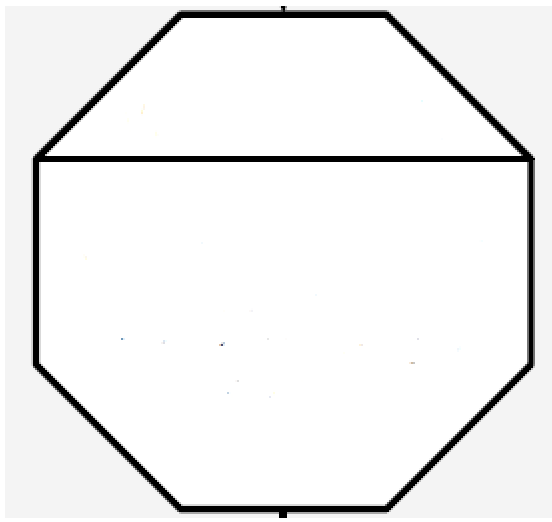
Comments: Radiological safety principles and procedures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	2.4.19	
	Importance Rating	3.4	

Knowledge of EOP layout, symbols, and icons.

Proposed Question: RO Question 73

What does this EOP Flowchart symbol indicate?



A. Decision Step

B. Hold/Wait Point

C. Instructional Step

D. Concurrent Execution

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This would be true if the shape was a diamond.

B. Correct: This shape is a Hold/Wait Point.

C. Incorrect: This would be true if the shape was a rectangle.

D. Incorrect: This would be true if the shape was a rectangle with subsequent steps for analysis and implementation.

Technical Reference(s):	EOP Bases – Flowchart Use Rev.13	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.01, interpret the meaning of the color and shape of any EOP flowchart step.	(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	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.29	
	Importance Rating	3.1	

Knowledge of the emergency plan.

Proposed Question: RO Question 74

What is the MINIMUM emergency event classification level in which the Emergency Response Organization (ERO) is required to be fully activated?

A. Notification of Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The ERO is not required to be activated for an Unusual Event. This is at the OSM's discretion.

B. Correct – The site ERO is activated and staffed in an emergency condition corresponding to an ALERT classification or higher.

C. Incorrect – The ERO would be activated for the levels but they are not the minimum.

D. Incorrect - The ERO would be activated for the levels but they are not the minimum.

Technical Reference(s): EPIP 1.3, Rev.19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.20 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	Monticello, 2013

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

10 CFR Part 55 Content:	55.41	10
	55.43	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.39	
	Importance Rating	3.9	

Knowledge of RO responsibilities in emergency plan implementation.

Proposed Question: RO Question 75

- You are the EANSOE assisting with rod sequence exchange
- An Unusual Event has been declared due to drywell leakage
- The OSM has determined that accountability will be taken

In accordance with the Emergency Plan, where are you required to go?

A. Control Room

B. Plant Access Building

C. Plant Support Center

D. Training Center Simulator

Proposed Answer: A

Explanation (Optional):

A. Correct – The control room is the location for on watch operating crew accountability.

B. Incorrect – Visitors to the site are required to be escorted to the Plant Access Building

C. Incorrect – Personnel in the Plant Support Center and fixed trailers outside the protected area not assigned to the ERO are to report to the Plant Support Center.

D. Incorrect – For Security related events per AOP 914 the Training Center Simulator is a designated area for command and control due to a loss of Control Room accessability.

Technical Reference(s): EPIP 1.3 Rev. 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: 95.00.00.20 (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	10
	55.43	

Comments: Administrative, normal, abnormal, and emergency operating procedures for the facility. **Security Related Information – Withhold From Public Disclosure**