EXAM COVER SHEET

EXAM RULES

Exam Number/Title: PDA OPS 19-1 NRC Exam, Rev. 0		**Trainee Name:**
Training Program: Operations		**Employee ID:**
Course/Lesson Plan Number(s): 50007		
Total Points Possible: 75	**PASS CRITERIA: ≥ 80%**	**Grade: _____/75=_____%**
Graded by:		**Date:**
EXAM REVIEW AND APPROVAL:		
Submitted by Instructor:		**Date:**
Technical Review by:		**Date:**
Approved by Training Supervision:		**Date:**

1. References may not be used during this exam, unless otherwise stated.
2. Read each question carefully before answering. If you have any questions or need clarification during the exam, contact the exam proctor.
3. Conversation with other trainees during the exam 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. Restroom trips are limited and only one examinee at a time may leave.
6. For exams with time limits, you have **480 minutes** to complete the exam.
7. The examinee agrees to refrain from discussing the content of the exam until the end of the exam cycle

EXAM 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 Exam Rules stated above. Further, I have not given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam."

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 exam 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 AK1.02	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Power/flow distribution.

Proposed Question: RO Question 1

The plant was operating at power when the “A” recirculation pump tripped due to a fault.

- Reactor power is 63%
- FR-4528 & PDR-4528 Total Core Flow / Core Plate Dp Recorder reads:
  - Total Core flow: 26 Mlbm/hr
  - Core Plate D/P: 3.6 psid

What action, if any, is required to be taken by the Operator?

A. insert control rods to lower power to approximately 58%

B. reduce recirculation speed to lower core flow to 25 Mlbm/hr

C. place the mode switch in shutdown

D. no further action is required

Proposed Answer: A

Explanation: AOP 255.2 For low core flow AND single loop operation conditions (i.e. <27 Mlbm/hr and/or < 7 psid), obtain Core Plate dp from PDR/FR-4528 and use Core Flow vs Core Plate dp graph under Attachment 2 to determine core flow in Mlbm/hr. Using 3.6 psid provided and graph core flow would be 22 Mlbm/hr. IPOI-3 plot on power to flow map using these values of core flow and power places the plant above the MELLA Limit.
AOP-264 Maintain the following administrative limits during single loop operation
Core power shall be less than or equal to 60%
Core Flow shall be less than or equal to 53%
Active loop jet pump flow shall be less than or equal to 32.0 Mlb/hr.

A. Correct
AOP 255.2 Rev 48. Follow up action 6.b determine core flow from Attachment 2 (22 Mlb/Hr. Plotting on IPOI-3 Power to flow map this places the unit above MELLA. Follow up ACTION 8 If inadvertent entry into area area above power to flow map (ie exceeding the 8 hour average load line of 100.64%) exit this area by inserting control rods. If unable to reduce power below the MELLA limit within one hour, manually scram the reactor. “

B. Incorrect
AOP 264 Rev 16 Follow up action 3, is required for single loop operation restrictions, however the first action would be to reduce core power to below the MELLA limit. 58% with control rods will establish this.
Based on plate dp / core flow plot plant is below 25 Mlbm/hr

C. Incorrect
Manual scram is not required unless the unit is not restored to below MELLA within 1 hour.

D. Incorrect, further action is required.

Technical Reference(s): AOP 264, Rev. 16
AOP 255.2 Rev 48
IPOI 3, Rev. 160
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: AOP 264, Att 1or AOP 255.2 Attachment 2 and IPOI 3 P/F map

Learning Objective: 12.01.01.01 Identify the appropriate procedures that govern the Recirculation System operation, include the operator responsibilities during all modes of operation, and any actions required by personnel outside of 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
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 G2.4.3	
	Importance Rating	3.7	

Partial or Complete Loss of AC Power: Ability to identify post-accident instrumentation.

Proposed Question: RO Question 2

Which of the following post-accident instruments would be unavailable if the RCIC Inverter failed?

A. "A" Wide Range Level Indicator LI 4539

B. "B" Wide Range Level Indicator LI 4540

C. "A" Fuel Zone Level Indicator LI 4565C

D. "B" Fuel Zone Level Indicator LI 4565B

Proposed Answer: A

Explanation:

A. Correct
AOP 302.1 Rev 59 Los of 1D13 probable indication 1C05 A Wide Range LI-4539 fails low
1D13 ckt 03 is RCIC Inverter

B. Incorrect: AOP 302.1 Loss of 125 VDC Power
1D23 probable indication 1C05 LI-4540 Wide Range fails low
OI-152A1Rev HPCI Electrical Lineup. 1D2302 is HPCI Inverter
OI152A1 HPCI Inverter is LT4540

C. Incorrect
LI4565C Fuel Zone is 1Y11power

D. Incorrect
LI4565B Fuel Zone is 1Y21 power

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

Proposed References to be provided to applicants during examination: N

Learning Objective:	5.01.01.02 Given a HPCI system operating mode and various plant conditions, predict how the HPCI system will be impacted by the following supported system failures: a. 125 VDC 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	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	6
	55.43	

(6) Design, components, and function of reactivity control mechanisms and instrumentation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AA2.03	
	Importance Rating	2.8	

295004 (APE 4) Partial or Total Loss of DC Power / 6: AA2.03 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Battery voltage. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: RO Question 3

The plant has experienced a station blackout and efforts are underway to restore power.

- 4 hours have elapsed
- RCIC is operating
- Reactor water level is stable

What is the concern with this prolonged loss of AC Power?

The extended loss of AC power results in a loss of ____________.

A. the RCIC suction path when it is realigned to the TORUS

B. all safety related DC power when the coping time of the station batteries is exceeded

C. the Reactor Coolant System boundary caused by excessive and prolonged use of the SRVs

D. containment vent capabilities when the environmental qualifications of containment are exceeded

Proposed Answer: B

Explanation:

A. Incorrect, RCIC suction swaps on Lo-Lo CST level

B. Correct, station batteries are sized to provide emergency power for a 4 hour time period

C. Containment vent capabilities EQ ratings are concerned with SRV solenoid valves

D. Loss of AC would not result initially cause a loss of RPS and a Group 1 isolation. However AOP 301.1 Station Blackout directs the plant cooldown at 80-100°F/Hr within 30 minutes. The commencement of the RPV cooldown will lower RPV pressure below the SRV setponts

Technical Reference(s):	SD 375 Rev 9 Plant DC Power System AOP 301.1 Rev 77 Station Blackout	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.01.05.06 Relate how each step and its performance meets the mitigation strategies of AOP 301.1 Station Blackout	(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	

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA1.01	
	Importance Rating	3.1	

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system: Plant-Specific.

Proposed Question: RO Question 4

The plant is operating at 50% when the Main Turbine TRIPS. The Operator is carrying out the IPOI 5, SCRAM actions and observes:

- both recirculating pumps running at 20% speed

What, if any, action(s) should have happened with reactor Recirc?

A. “B” recirculation pump trips

B. both recirculation pump speeds reduce to minimum

C. both recirculation pumps trip

D. no additional actions

Proposed Answer: C

Explanation:

A. Incorrect.
Automatic action failed to occur, trip both Recirc pumps

B. Incorrect
At >26% power with a Turbine Trip RPT breakers will trip (Recirc pumps 1P201A &B both trip). 20% is minimum speed

C. Correct
Recirc pumps should have auto tripped via RPT on the Turbine Trip at >26% power

D. Incorrect.
Automatic action failed to occur, trip both Recirc pumps

Technical Reference(s): ARP 1C05B (A-5) Rev 108 RPT System A or B Trip
IPOI-5 Rev 62 Appendix 2 Case 2 Scram caused by turbine trip
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 12.01.01.12 Describe the effect on RPV water level, that is caused by Recirculation Pump operations (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

(7) 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 #	295006 AK3.01	
	Importance Rating	3.8	

295006 (APE 6) Scram / 1: AK3.01 - Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor water level response. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 5

Which of the following is the expected initial response of reactor water level when a manual reactor scram is inserted from 100% reactor power and which system operation impacts this response?

Reactor water level will **INITIALLY** ______________.

A. rise due to the delayed response of the feed regulating valves

B. rise due to the rise in CRD cooling water flow

C. lower due to main turbine bypass valves opening

D. lower due to control rod insertion

Proposed Answer: D

Explanation: IPOI-5 Rev 62 Appendix 2 Automatic Actions Case 1 Reactor Scram NOT initiated by turbine trip or main steam isolation. Step 3 The Condensate and reactor feed pump recirculation valves open and the feedwater control system attempts to maintain level. Reactor water level will initially shrink due void collapse.

A. Incorrect
RPV/L will initially lower due to control rod insertion and void collapse

B. Incorrect
RPV/L will initially lower due to control rod insertion and void collapse

C. Incorrect
RPV/L will initially lower due to control rod insertion and void collapse.
Turbine BPV opening will lower RPV pressure and will cause level rise on lowering pressure and swell

D. Correct
RPV/L will initially lower due to control rod insertion and void collapse.

Technical Reference(s): IPOI-5 Rev 62 Appendix 2 Automatic Actions (page 13 of 15) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 93.22.01.02 Differentiate beween the plant response to a scram from a Turbine Trip, MSIV isolation, and any non-turbine related scram 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 5

55.45 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK2.02	
	Importance Rating	4.0	

295016 (APE 16) Control Room Abandonment / 7: AK2.02 - Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific. (CFR: 41.7 / 45.8)

Proposed Question: RO Question 6

After control room abandonment, with RCIC in operation per AOP-915 (Shutdown Outside Control Room)

How will RCIC respond if RPV level rises above 211 inches?

RCIC will ________.

A. continue to inject

B. ISOLATE and remain isolated

C. TRIP and remain in a TRIP condition

D. shutdown and will automatically restart

Proposed Answer: D

Explanation: ARP 1C-04C (A-6) Auto Action, High Level Trip MO-2404 Turbine Steam Supply valve closes, the steam supply valve closure will auto clear at Reactor Lo-Lo Level of 119.5" and MO-2404 will reopen.

A. Incorrect
MO2404 will auto close on high level 211 inches, however will automatically re-open at low-low of 119.5 inches

B. Incorrect
RPV 211 inches is a trip signal to MO-2404

C. Incorrect
RPV 211 inches is not an isolation signal

D. RSP Transfer HS2130 Transfers LI4540 & Auto Level. Transfers source of RPV high level trip / shutdowns to LITS-4540 HPCI,RCIC, RFPs main turbine.
MO2404 RCIC Steam Supply. The valve automatically opens on a RCIC system initiation signal. The valve automatically closes at an RPV water level of 211 inches.

Technical Reference(s): SD-925 Rev 8 Remote Shutdown Panel
SD-150 Rev 9
(Attach if not previously provided)

ARP 1C-04C (A-6)

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	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	7
	55.45	8

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

295018 (APE 18) Partial or Complete Loss of CCW / 8: AK1.01 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations. (CFR: 41.8 to 41.10)

Proposed Question: RO Question 7

With the plant operating at 100% reactor power what action, if any, would an Operator take for a complete loss of RBCCW to the drywell that lasts for 15 minutes?

A. insert a manual reactor scram

B. raise CRD mini-purge flow rate to 5 GPM

C. lower recirculation pump speed to minimum

D. no Operator action is required for this time period

Proposed Answer: A

Explanation: ARP 1C-06B (D-3) Operator Action Step 3.7 If continued cooling to the Reactor Recirc Pumps is not possible, manually scram the reactor per IPOI-5 (Reactor Scram) and secure the Reactor Recirc Pumps within 10 minutes or if seal cavity temperatures reach 250°F TR-4600 on 1C21.
OI-264 P&L 7, Do not Operate recirc pumps more than 10 minutes without RBCCW cooling water to the pump seals.

A. Correct
ARP 1C06B (D-3) RBCCW Pump Discharge Low Pressure
If continued cooling of the Reactor Recirc Pumps is not possible, manually scram the reactor per IPOI-5 Reactor Scram and secure Reactor Recirc Pumps within 10 minutes or if seal cavity temperatures reach 250°F TR4600 on 1C21.

B. Incorrect
Loss of RBCCW cooling to requires scram and securing Recirc pumps within 10 minutes

C. Incorrect
Loss of RBCCW cooling to requires scram and securing Recirc pumps within 10 minutes

D. Incorrect
Loss of RBCCW cooling to requires scram and securing Recirc pumps within 10 minutes

Technical Reference(s):	ARP 1C06B (D-3) OI-264 Rev 143 P&L 7 (page 4 of 65)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	29.01.01.02 Identify the appropriate procedures that govern the RBCCW system operation, include 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	8-10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 G2.1.7	
	Importance Rating	4.4	

(APE 19) Partial or Complete Loss of Instrument Air: G2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Proposed Question: RO Question 8

The plant is operating at 50% reactor power when a complete loss of plant air occurs.

The following indications are received in the Control Room:

- Rod Drift Annunciator
- Scram Air Header HI/Low Pressure Alarm

What Operator action should be taken to address this condition?

A. Scram the reactor immediately

B. Continue to monitor for any additional drifting control rod

C. Place the Emergency Rod In switch to EMERGENCY IN momentarily

D. Run an ACUMEN Report to determine if thermal limits remain acceptable

Proposed Answer: A

Explanation: AOP-518 Rev 42 immediate actions require manually scramming the reactor if instrument air header pressure is rapidly decreasing or cannot be restored or any rod has drifted
1C-05B (F-1) Scram Air Header HI/LO Pressure, Operator Action, If any control rod starts drifting into the core, manually scram the reactpr per IPOI-5 (Reactor Scram)

A. Correct
AOP-518 Rev 42 immediate actions require manually scramming the reactor if instrument air header pressure is rapidly decreasing or cannot be restored or any rod has drifted

B. Incorrect,
This condition requires a manual scram

C. Incorrect,
This condition requires a manual scram

D. Incorrect,
This condition requires a manual scram

Technical Reference(s):	AOP-518 Rev 42 1C-05B (F-1) Rev 108 (page 103 of 113)	(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 indication 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	5
	55.43/45	5/12-13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AA2.01	
	Importance Rating	3.5	

295021 (APE 21) Loss of Shutdown Cooling / 4: AA2.01 - Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water heatup/cooldown rate. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: RO Question 9

The plant is in MODE 4 with Shutdown cooling in service on the "A" side. The "B" Reactor Recirc Pump is running at minimum speed. The following occurs:

- Essential 4160V Bus 1A3 Lockout

(1) Which one of the following describes the automatic actions that occur as a result of these conditions?

AND

(2) How is coolant temperature monitored?

A. (1) The "B" side Shutdown Cooling system will automatically align to cool the reactor
(2) Coolant temperature is monitored via RWCU drain temperature

B. (1) The "B" side Shutdown Cooling system will automatically align to cool the reactor
(2) Coolant temperature will be monitored via the Reactor Recirc loop temperature

C. (1) Shutdown cooling will be lost and requires manual alignment to place "B" side in service
(2) Coolant temperature is monitored via RWCU drain temperature

D. (1) Shutdown cooling will be lost and requires manual alignment to place "B" side in service
(2) Coolant temperature will be monitored via the Reactor Recirc loop temperature

Proposed Answer: D

Explanation: Loss of 1A3 will cause a loss of A Side RPS. Group 4 PCIS logic is A RPS powered. MO-1908 SDC inboard suction will close. MO-2003 A LPCI inject will close if MO-1908 and MO-1909 were open.

M)-1908 is powered by 1B3438 and will be lost on the 1A3 lockout
MO-2003 is powered via 1B3493 and will maintain power via 1B34A/44A LPCI swing Bus.
MO-1908 will require manual alignment to be re-opened.
.

A. Incorrect
No automatic realignment to the B side SDC. Loss of 1A3 will cause loss of A side RPS and Group Isolations 1 thru 5. (Group 5 will isolate RWCU)
Loss of 1A3 will cause the loss of A RHR and RHRSW pumps

B. Incorrect
No automatic realignment to the B side SDC. Loss of 1A3 will cause loss of A side RPS and Group Isolations 1 thru 5. (Group 5 will isolate RWCU)
Loss of 1A3 will cause the loss of A RHR and RHRSW pumps
AOP-149 Loss of Shutdown Cooling directs manually opening MO-1908 / 1909, MO1905 / 2003 to re-establish SDC

C. Incorrect
No automatic realignment to the B side SDC. Loss of 1A3 will cause loss of A side RPS and Group Isolations 1 thru 5. (Group 5 will isolate RWCU)
Loss of 1A3 will cause the loss of A RHR and RHRSW pumps

D. Correct
No automatic realignment to the B side SDC. Loss of 1A3 will cause loss of A side RPS and Group Isolations 1 thru 5. (Group 5 will isolate RWCU)
Loss of 1A3 will cause the loss of A RHR and RHRSW pumps

Technical Reference(s):	AOP-149 Rev 147 Loss of Shutdown Cooling AOP 301 Rev 74, Loss of Essential Electrical Power AOP-358 Rev 32 (page 2 Auro Actions) ARP 1C05B (D-4) Rev 108 (page 80 of 113)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	5.01.01.01 Evaluate plant conditions and control room indications and determine the actions directed by AOP 149	(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	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA1.06	
	Importance Rating	3.3	

295023 (APE 23) Refueling Accidents / 8: AA1.06 - Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Neutron monitoring. (CFR: 41.7 / 45.6)

Proposed Question: RO Question 10

The MODE switch is in REFUEL with the vessel head removed.

The following SRM indications are noted:

	Period	CPS
SRM A	∞	1E3
SRM B	∞	1E3
SRM C	+10 (steady)	2E4
SRM D	+100 (steady)	5E5

What signals are automatically generated by the SRMs, if any?

A. Rod Block only

B. Half Scram and Rod Block

C. Full Scram and Rod Block

D. No automatic signals are generated

Proposed Answer: A

A
SRM channel upscale neutron flux level trip was only used during initial fuel loading and low power physics testing to provide reactor protection until overlap between the SRM and IRM s was demonstrated. Any single SRM, IRM, or APRM upscale or IRM or APRM Inop trip would have produced a trip signal in both channels A3 and B3 of RPS

A. 1E5 cps is the SRM Upscale setpoint.
Rod Block only

B. 1E5 cps is the SRM Upscale setpoint.
Only a Rod Block
Shorting Links installed preventing RPS trip

C. 1E5 cps is the SRM Upscale setpoint.
Shorting Links installed preventing RPS trip

D. 1E5 cps is the SRM Upscale setpoint.
Shorting Links installed preventing RPS trip

Technical Reference(s):	SD358 Rev 9	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	78.02.01.06 Describe the SRM system interlocks (include alarms), 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
	55.45	6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK3.07	
	Importance Rating	3.5	

295024 High Drywell Pressure / 5: EK3.07 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Drywell venting. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 11

Primary Containment Pressure is rising following a large RCS break inside the primary containment. Primary Containment parameters are as follow:

- Drywell Pressure indication is 49 psig and slowly rising
- Torus pressure indication is 45 psig and slowly rising
- Wide Range Torus level indication is 12.5 feet and stable

IAW EOP 2, Primary Containment Control,

(1) What is the correct vent path?

AND

(2) why?

A. (1) Torus vent through CV 4301 OUTBD TORUS VENT ISOL, CV 4309 INBD TORUS VENT BYPASS ISOL and CV 4300 INBD TORUS VENT ISOL as required
(2) The filtering of the SBGT system will reduce the offsite release rate

B. (1) Drywell vent through CV 4303 OUTBD DRYWELL VENT ISOL, CV 4310 INBD DW VENT BYPASS ISOL and CV 4302 INBD DRYWELL VENT ISOL as required
(2) This is the ONLY vent path available due to level in the TORUS

C. (1) Drywell vent through CV 4303 OUTBD DRYWELL VENT ISOL, CV 4310 INBD DW VENT BYPASS ISOL and CV 4302 INBD DRYWELL VENT ISOL as required
(2) This vent path will release to the Offgas Stack for an elevated release dilution

D. (1) Torus vent through the hardened vent path via CV-4360, TORUS HARDPIPE VENT INBOARD ISOLATION and CV 4361 TORUS HARD PIPE VENT OUTBD ISOLATION as required
(2) This path eliminates the potential for duct work or SBGT failure during venting which would significantly increase radioactivity levels in the reactor building.

Proposed Answer: A

Explanation:

A. Correct:
Torus venting is normally preferred due to the scrubbing provided by the torus and this path also utilizes SBGT.

B. Torus venting is normally preferred due to the scrubbing provided by the torus and this path also utilizes SBGT
Torus level is below 16 ft. The level at which the Drywell is vented

C. Torus venting is normally preferred due to the scrubbing provided by the torus and this path also utilizes SBGT
SEP 301.3 Torus Vent via Hardpipe vent is performed when the normal Torus vent path is unavailable. In this case normal Torus vent is available.
Torus level is below 16 ft. The level at which the Drywell is vented

D. Torus level is below 16 ft. The level at which the Drywell is vented
Torus venting is normally preferred due to the scrubbing provided by the torus and this path also utilizes SBGT

Technical Reference(s):	SEP 301.1 Torus Vent via SBGT Rev 13 Sep 301.3 Torus Vent via Hardpipe Vent Rev 15 EOP-2 Rev 18, Step PC/P-10	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.64.16.01 Contrast the effects of venting the drywell and venting the torus	(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
	55.45	6

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

295025 (EPE 2) High Reactor Pressure / 3: EK2.08 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system: Plant- Specific. (CFR: 41.7 / 45.8)

Proposed Question: RO Question 12

The plant was operating at 95% power with a shutdown in progress.

- The pressure transmitter associated with the “A” pressure regulator is mechanically bound such that its output signal does **NOT** change

As power is further reduced, the “B” pressure regulator will ________.

A. assume control and reactor pressure will be controlled 5 psig higher

B. assume control and no changes in plant response will be noted

C. NOT assume control and the reactor will receive an RPS trip on high pressure

D. NOT assume control and the reactor will receive an RPS trip on MSIV closure

Proposed Answer: D

Explanation:
Pressure Regulator Signal Failure Low: backup pressure regulator controls pressure 5 psig higher
Pressure Regulator Signal Failure High: Reactor pressure lowers until a Group 1 isolation (850 psig) results in a reactor scram. Group 1 pressure is measured at the main steam averaging manifold, PR-1000 Main Steam Pressure Recorder on 1C07 is used to access the Group 1 Isolation

A. Incorrect,
As the “A” side pressure error signal goes down, the “B” pressure error signal will go up. This will eventually cause the “B” regulator to take over and control pressure slightly higher (approx. 5 psig)
No scram will occur

B. Incorrect,
As the “A” side pressure error signal goes down, the “B” pressure error signal will go up. This will eventually cause the “B” regulator to take over and control pressure slightly higher (approx. 5 psig). This failure will not cause the 850 psig Group 1
No scram will occur

C. Incorrect,

As the "A" side pressure error signal goes down, the "B" pressure error signal will go up. This will eventually cause the "B" regulator to take over and control pressure slightly higher (approx. 5 psig).

D. Correct
As the "A" side pressure error signal goes down, the "B" pressure error signal will go up. This will eventually cause the "B" regulator to take over and control pressure slightly higher (approx. 5 psig).

Technical Reference(s): AOP-262 Rev 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 52.01.01.02 Given an EHC system operating mode and various plant conditions, predict how the EHC system will be impacted by failures in the following support systems:
a. 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
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.45 8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EK1.02	
	Importance Rating	3.5	

295026 (EPE 3) Suppression Pool High Water Temperature / 5: EK1.02 - Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Steam condensation. (CFR: 41.8 to 41.10)

Proposed Question: RO Question 13

The plant is operating at 100% reactor power.

- Suppression pool water temperature is 90°F and rising

The first operational limit is reached based upon the Suppression Pool having capability to provide…

A. Adequate Net Positive Suction Head to the ECCS pumps

B. Adequate heat capacity in the Torus to prevent exceeding HCTL during an ATWS

C. Complete steam condensation following a Loss of Coolant Accident

D. Complete steam condensation following inadvertent Safety Relief Valve actuation

Proposed Answer: C

Explanation: The first operational limit is 95°F which is the assumed initial temperature of the LOCA analysis. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBA's) The suppression must quench all the steam released through the downcomer lines during a Loss of Coolant Accident (LOCA) TS Bases 3.6.2.1 The technical concerns that lead to the development of the suppression pool average temperature limits are as follows: a) complete steam condensation, the original limit for the end of a LOCA blowdown was 170 °F based on the Bodega Bay and Humboldt Bay Tests. b) Primary containment peak pressure and temp design pressure is 56 psig and design temp is 281°F

A. Incorrect NPSH is not adversely affected at 90°F Torus water temperature

B. Incorrect, heat capacity limit is not adversely affected by 90°F Torus water temperature. Typically 160°F is where capacity limit is approached or exceeded

C. Correct The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBA's) The suppression must quench all the steam released through the downcomer lines during a Loss of Coolant Accident (LOCA) TS Bases 3.6.2.1 The technical concerns that lead to the development of the suppression pool average

temperature limits are as follows: a) complete steam condensation, the original limit for the end of a LOCA blowdown was 170 °F based on the Bodega Bay and Humboldt Bay Tests. b) Primary containment peak pressure and temp design pressure is 56 psig and design temp is 281°F

D. Incorrect

Technical Reference(s): Tech Bases 3.6.2.1 Amendment 223 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.15 Explain the Bases of each of the EOP Curves and Limits (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 8, 10

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 G2.4.46	
	Importance Rating	4.2	

295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5: Generic K/A 2.4.46 - Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Proposed Question: RO Question 14

The plant is operating at 100% reactor power.

- All well water cooling to the drywell is lost

Which one of the following PCIS Isolations would be received as a result of no Operator actions?

A. Group 1

B. Group 2

C. Group 5

D. Group 7

Proposed Answer: B

Explanation: Loss of well water cooling to the drywell will cause drywell pressure to rise and with no operator action to vent the drywell 2 psig signal will occur. 2 psig is a Group 2 PCIS isolation signal.

A. A loss of WW cooling to the DW will cause DW pressure to rise. 2 psig is a PCIS setpoint. Group 1 does not have a 2 psig setpoint / signal

B. Correct the PCIS Group 2 does have a 2 psig setpoint / signal.

C. A loss of WW cooling to the DW will cause DW pressure to rise. 2 psig is a PCIS setpoint. Group 5 does not have a 2 psig setpoint / signal

D. A loss of WW cooling to the DW will cause DW pressure to rise. 2 psig is a PCIS setpoint. Group 7 does not have a 2 psig setpoint / signal

Technical Reference(s):	SD 959.1 Rev 13 Primary Containment Isolation System ARP 1C05B (A-1) Rev 108 Primary Containment Hi Pressure Trip AOP-408 rev 34	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	26.01.01.13 Given a well water system operating mode and various conditions, predict how each supported system will be impacted by the following well water system failures a. One or more Well Water Pump(s) trip	(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
	55.43/45	5/5,12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EA2.01	
	Importance Rating	4.1	

295030 (EPE 7) Low Suppression Pool Water Level / 5: EA2.01 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool level. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: RO Question 15

Which of the following describes the level and reason for performing an Emergency Depressurization on Torus Water Level low?

Torus water level cannot be maintained above ________.

A. 5.8 Feet, to ensure steam discharged from the drywell to the Torus following a primary system break will be adequately condensed

B. 5.8 Feet, to ensure opening an SRV will not result in exceeding the code allowable stresses in the SRV Tailpipe

C. 7.1 Feet, to ensure steam discharged from the drywell to the Torus following a primary system break will be adequately condensed

D. 7.1 Feet, to ensure opening an SRV will not result in exceeding the code allowable stresses in the SRV Tailpipe

Proposed Answer: C

Explanation: Torus level of 7.1 Ft corresponds to the bottom of the drywell to torus downcomers. Torus level below 7.1 ft could result in loss of the pressure suppression function of the primary containment

A. 5.8 FT Corresponds to the HPCI turbine exhaust elevation. Operation of HPCI with its exhaust unsubmerged will tend to directly pressurize the torus

B. 5.8 FT Corresponds to the HPCI turbine exhaust elevation. Operation of HPCI with its exhaust unsubmerged will tend to directly pressurize the torus

C. Correct
7.1 A Torus level of 7.1 Ft corresponds to the bottom of the drywell to torus downcomers. Torus level below 7.1 ft could result in loss of the pressure suppression function of the primary containment.

D. 7.1 A Torus level of 7.1 Ft corresponds to the bottom of the drywell to torus downcomers. Torus level below 7.1 ft could result in loss of the pressure suppression function of the primary containment.

Technical Reference(s): EOP-2 Rev 16 Bases page 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.15 Explain the Bases of each of the EOP Curves and Limits (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

55.43/45 5/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EA1.05	
	Importance Rating	4.3	

295031 (EPE 8) Reactor Low Water Level / 2: EA1.05 - Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Reactor core isolation system. (CFR: 41.7 / 45.6)

Proposed Question: RO Question 16

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

- Reactor water level is 110 inches and lowering
- Drywell pressure is 2.2 psig and stable

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

- Reactor Core Isolation Cooling (RCIC) is in a standby readiness condition

Is this expected for the given plant conditions and what action is required?

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

C. No, RCIC 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 RCIC

D. No, RCIC should have receive an auto start signal if 1C05B (A-1), Primary Containment HI Pressure Trip, annunciator is received and the BOP should manually start RCIC

Proposed Answer: C

Explanation: RCIC should have automatically started when reactor water level lowered to the Reactor Lo-Lo Level Trip of 119.5”. EOP-1 RC-4 Initiate any of the following which should have initiated but did not: (Isolations, ECCS Initiations, SBDG initiations)

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. RCIC should have automatically started when reactor water level lowered to Reactor LO-LO Level Trip.

C. Correct: RCIC 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 RCIC.

D. Incorrect: The Primary Containment High Pressure signal of 2 psig is a HPCI automatic start. RCIC does not have a Primary Containment High Pressure start. HPCI does also have the LO-LO Reactor Level 119.5” auto start signal..

Technical Reference(s): 1C05A (B-1) Rev 90 Reactor Water LO-LO Level Trip 1C05B (A-1) Rev. 108 Primary Containment Hi Pressure Trip (Attach if not previously provided)

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

55.45 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK3.02	
	Importance Rating	4.3	

295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1: EK3.02 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC injection. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 17

The plant has experienced an ATWS. The following conditions are present:

- Reactor Power is 25%
- Reactor Water Level is 158 inches and stable
- Reactor Pressure is 1000 psig and stable
- ARI was unsuccessful shutting down the reactor
- Torus Water Temperature is 97°F and rising
- MSIVs are closed

The RO reports "The ATWS QRC is complete with the exception of Boron Injection"

(1) Should Boron be injected?

AND

(2) Why?

A. No, Boron injection should wait until the Boron Initiation Injection Temperature is exceeded to prevent excessive cleanup efforts following the event.

B. No, Boron injection should await intentionally lowering RPV water level to promote homogenous mixing of the Boron solution minimizing thermal hydraulic instabilities.

C. Yes, Boron injection should occur to prevent uneven distribution of boron in the core resulting in inadequate shutdown margin and localized criticality.

D. Yes, Boron injection should occur prior to exceeding the Boron Initiation Injection Temperature to prevent exceeding the Heat Capacity Limit of the Torus.

Proposed Answer: D

Explanation:

A. Incorrect, reactor is at 25% power and Torus temperature is rising.

B. Incorrect: Maintaining RPV water level low in the expanded band may reduce boron mixing efficiency and delay reactor shutdown/ (ATWS Bases Rev 19, page 41)

C. Incorrect, uneven distribution of boron in the core is not a concern at this level. At 25% and greater than 87” in RPV power level control will be entered and lowered intentionally to choke reactor power.

D. Correct: The boron injection requirement is established in the Steps Q-6 and Q-7 (before torus water temperature reaches the BIIT) If boron injection is required, heat is being added to the containment and emergency depressurization may be required. The second condition, found in Step / Q-6, is based on inserting boron before exceeding the Heat Capacity Limit due to high torus water temperature. (page 91 of 96).
A SCRAM failure coupled with an MSIV isolation, however results in rapid heatup of the torus due to steam discharged from the RPV via SRVs. The challenge to the primary containment thus becomes the limiting factor which defines the requirement for boron injection.

Technical Reference(s): ATWS bases Rev 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.15 Explain the Bases of each of the EOP Curves and Limits (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

55.45 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EK2.07	
	Importance Rating	3.5	

295038 (EPE 15) High Offsite Radioactivity Release Rate / 9: EK2.07 - Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Control room ventilation. (CFR: 41.7 / 45.8)

Proposed Question: RO Question 18

Standby Filter Units have actuated on a valid signal. RO investigates and finds that “A” and “B” Battery room exhaust fans are operating.

What action(s) should the Operator take?

A. Trip only the “A” Battery Room Exhaust fan

B. Trip only the “B” Battery Room Exhaust fan

C. Trip all running Battery Room Exhaust fans

D. Start all non-running Battery Room Exhaust fans

Proposed Answer: A

Explanation:

A. Correct: OI-730 after an isolation only 1V-EF-30B or C shall be running. If flow from the battery rooms is not limited to 100 cfm following an isolation, the control room SFU may not be able to maintain the control room at a positive pressure. Therefore to maintain positive pressure during a Control Building Isolation, only one Battery Exhaust Fan 1V-EF-30B or 1V-EF-30C shall be running
1V-EF-30A capacity is 700 CFM
1V-EF-30B and C capacity is 100 CFM
In a normal configuration 1V-EF-30A is normally running along with the B or C fan.

B. Incorrect

C. Incorrect

D. Incorrect

Technical Reference(s): OI 730, Rev. 127 P&L 8 page 4 of 67
Page 52 of 67 Section 6.1 Automatic Startup of SFU
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	65.01.01.01 Relate the precautions and limitation, operating cautions, or procedural notes of OI 730 to any component or Control Building HVAC system operating status	(As available)

Question Source:	Bank #	STP 3.0.0-02 Rev 59 (page 22 or 31)Control Room Panel Checks SD 730 Rev 12	
	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
	55.45	8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AK1.02	
	Importance Rating	2.9	

Plant Fire On Site - Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting.

Proposed Question: RO Question 19

The plant is operating at 100% reactor power when facilities personnel report an active fire in the “A” RFP motor.

The following alarms are now received at 1C40:

- 1C40 C-3 “A” RFP 1P-1A Deluge #4 Initiated
- 1C40 J-5 Electric Fire Pump 1P-48 Running

In accordance with AOP 913, is offsite assistance required to be requested?

A. Yes, because the fire is within the power block

B. Yes, because water is needed to extinguish the fire

C. No, because the fire is not on safety related equipment

D. No, because “A” RFP 1P-1A Deluge #4 is sufficient to extinguish the fire

Proposed Answer: B

Explanation:

A. Incorrect: AOP does not limit offsite assistance to fires within the protected area only

B. Correct Step 6 of the immediate actions states, “If water is required to extinguish OR if fire is outside the protected area, THEN request offsite fire assistance per OFFSITE ASSISTANCE section and request additional Fire Brigade support.” Active fire reported and 1P048 Electric Fire Pump is running and A RFP deluge has initiated.

C. Incorrect: AOP does not limit offsite assistance to only safety related

D. Incorrect: Active fire reported. Although the deluge is initiated, additional offsite assistance should be requested.

Technical Reference(s): AOP-913 Rev 83 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.25.01.03 Given any fire alarm or when notified by plant personnel, evaluate plant conditions and control room indication to determine the required operator actions in accordance with AOP 913	(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	

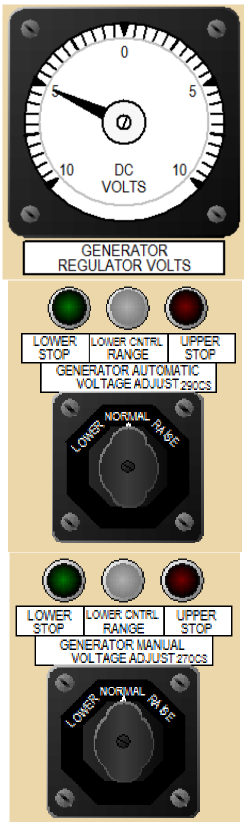
10 CFR Part 55 Content:	55.41	7
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000	G2.4.31
	Importance Rating	4.2	

Generator Voltage and Electric Grid Disturbances - Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: RO Question 20

The plant is operating at 100% reactor power when a grid transient occurs which resulted in the following indication:



To null the meter, the Operator will place the Generator ____(1)____ Voltage Adjust Control Switch in the ___(2)___ direction?

A. (1) Manual
(2) LOWER

B. (1) Manual
(2) RAISE

C. (1) Automatic
(2) LOWER

D. (1) Automatic
(2) RAISE

Proposed Answer: A

Explanation: At 100% Auto Voltage Regulator is normally in service. The Manual Regulator should be periodically adjusted so that the Generator Regulator Volts meter indicates zero Typically the adjustment is towards the direction the meter is deflected from zero. In this case towards lower

A. Correct: Section 4.2 Periodic Checks step 4 Voltage regulation should remain in automatic. The Manual Regulator should be periodically adjusted so that the Generator Regulator Volts meter indicates zero.

B. Incorrect

C. Incorrect

D. Incorrect

Technical Reference(s): OI 698, Rev. 103 (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 necessary to place the main generator 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	10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295008 AA2.05	
	Importance Rating	2.9	

High Reactor Water Level - Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Swell. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: RO Question 21

The plant is operating at 100% reactor power.

- "B" Reactor Recirc MG set spuriously runs back to 45% speed

Which one of the following describes the RPV level response to this malfunction?

Indicated RPV Water Level _________.

A. lowers and the Reactor scrams on low level

B. lowers and the Reactor does **NOT** scram on low level

C. rises and the Main Turbine TRIPs on high level

D. rises and the Main Turbine does **NOT** TRIP

Proposed Answer: D

Explanation: 45% runback will cause power to lower and less steam from the reactor. The feedwater control system will sense the rising water level and automatically adjust to close the feedwater reg valves and control reactor level

A. Incorrect, on runback level will rise and may come close to 211' reactor feed pump and turbine trip setpoints. RPV level will not lower to the low scram setpoint

B. Incorrect level dose not lower it will rise

C. Incorrect level will rise and feedwater control system will sense the high level and error signal will close the feedwater reg valves.

D. Correct

Technical Reference(s): SD-644 Rev 16 page 42 Figure 5 Feedwater Control System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	45.02.01.04 Given a Feed and Condesate System operating mode and various plant conditions, predict how each supported system will be impacted by railures in the feed and condensate system: F. Recirculation 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
	55.43/45	5/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295012 AA1.01	
	Importance Rating	3.5	

High Drywell Temperature - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell ventilation system.

Proposed Question: RO Question 22

The plant was operating at 100% power with the Drywell Cooling system in its normal full power configuration when a small steam leak in the Drywell occurred.

The following is the sequence of events:

- EOP 2 is entered when Drywell temperature exceeds 150°F
- Drywell pressure exceeds 2 psig
- Drywell spray was initiated to maintain Drywell temperature < 280°F

Assuming no other operator action has been taken, which of the following is correct regarding the automatic response of the drywell cooling fans to the above sequence?

All running fans __________.

A. tripped when Drywell pressure exceeded 2 psig and no other automatic action occurred

B. shifted to slow speed when Drywell pressure exceeded 2 psig and no other automatic action occurred

C. shifted to slow speed when Drywell pressure exceeded 2 psig and all fans tripped when Drywell spray was initiated

D. remained at their original speed when Drywell pressure exceeded 2 psig and all fans tripped when Drywell spray was initiated

Proposed Answer: C

Explanation:

A. Incorrect: Fans shift to slow speed when drywell pressure exceeds 2 psig. Additionally the fans tripped when drywell spray was initiated sprays are initiated.

B. Incorrect: Drywell fans all tripped once drywell spray was initiated.

C. Correct: All fans shift to slow speed when drywell pressure exceeds 2 psig. All fans trip when drywell sprays are initiated.

D. Incorrect: All drywell fans are normally running in fast speed. Fans shift to slow speed when drywell pressure exceeds 2 psig. Additionally all drywell fans trip when drywell

Technical Reference(s): SD760 Rev 8 page 10, 14 (Note 2)
Figure 2 page 18 logic trips fan with Drywell Spray valves open (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 2.01.01.07 Given an RHR system operating mode and various plant conditions, predict how each supported system will be impacted by the following RHR system operations/failures:
c. Containment Spray Initiation (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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014 AK3.02	
	Importance Rating	3.7	

Inadvertent Reactivity Addition - Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: Control rod blocks.

Proposed Question: RO Question 23

The Rod Block Monitor System is used to prevent?

A. creating a high worth control rod notch

B. withdrawing a control rod during refueling

C. inserting a control rod out of its step sequence

D. violating the MCPR limit for a single rod withdraw error

Proposed Answer: D

Explanation:

A. Incorrect: This is minimized by development of the control rod sequence design.

B. Incorrect: This is a function of the Rod Out Block function of the Reactor Manual Control System.

C. Incorrect: This is a function of the Rod Worth Minimizer to enforce the control rod sequence.

D. Correct: RBM uses LPRM outputs to determine the thermal power production in a localized area around any selected rod. This information is utilized to block control rod withdraw that could result in a violation of Safety Limit Minimum Critical Power Ratio (SLMCPR) from a single rod withdraw.

Technical Reference(s): SD878.5 RBM Rev 10(Purpose page 4) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 82.01.01.04 State when the Rod Block monitoring system is required to be (As available)

operable by technical specifications
and describe the bases of the Rod
Block monitoring system LCOs

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	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020 AK2.11	
	Importance Rating	3.2	

Inadvertent Containment Isolation - Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Standby gas treatment system/FRVS: Plant-Specific.

Proposed Question: RO Question 24

The Fuel Pool Exhaust RIS-4131B Rad Monitor has failed with an INOPERABLE signal.

See drawing on the next page

Is this an expected response for this failure?

A. No, the “A” Train of SBGT should have started

B. No, the “B” Train of SBGT should have started

C. No, both trains of SBGT should have started

D. Yes, an INOPERABLE signal does not start SBGT

Proposed Answer: B

Explanation:

A. Incorrect: An INOPERABLE signal created by the Radiation Monitor will cause start signal to the “B” Standby Gas Treatment System (SBGT)

B. Correct: An INOPERABLE signal created by the Radiation Monitor will cause start signal to the “B” Standby Gas Treatment System (SBGT). From the indications provided in the question, the SBGT train remained in Standby.

C. Incorrect: The “A” SBGT train would receive a low flow autostart signal if it had received a start signal and was manually placed in Standby by an Operator.

D. Incorrect: An INOPERABLE signal created by the Radiation Monitor will cause start signal to the “B” Standby Gas Treatment System (SBGT)

Technical Reference(s): 1C03A (C-1) Fuel Pool Exhaust RIS-4131A/B Rad Monitor Dnscl/Inop Rev 63; 1C05B (C-8) PCIS Group 3 Isolation Initiated Rev 108 (Attach if not previously provided)

OI-170 Section 4.1 Automatic Initiation of SBGT Rev 66 (page 8 &9)

Proposed References to be provided to applicants during examination: N

Learning Objective:	7.00.00.02 Given a SBGT system operating mode and various plant conditions, predict how the SBGT system will be impacted by failures in the following support systems: a. PCIS	(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	

1C24


ABANDONED IN PLACE
ANNUNCIATOR
THESE SBGT CONTROLS ARE FUNCTIONAL ONLY IN MANUAL MODE

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EK1.01	
	Importance Rating	3.4	

295029 (EPE 6) High Suppression Pool Water Level / 5: EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Containment integrity. (CFR: 41.8 to 41.10)

Proposed Question: RO Question 25

The bases for maintaining Torus level below 13.5 feet is to prevent ________.

A. a loss of Torus level indication

B. covering the DW-to-Torus vacuum breakers

C. over-pressurizing the Torus with HPCI running

D. over-pressurizing the Torus with an SRV open

Proposed Answer: B

Explanation: Initially Torus level is maintained below the elevation of the bottom of the torus to drywell vacuum breakers to preserve the operability of these valves and thereby permit operation of drywell sprays. These vacuum breakers will not function as designed if any portion of the valve is covered with water. Keeping torus level below 13.5 ft assures that no portion of the drywell side of the valve is submerged

A. This would be 5.8 ft and uncovering the HPCI steam exhaust

B. 13.8 feet is the SRV Tailpipe Level Limit

C. 16 ft is upper tap, no concern with lower tap

D. Correct

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

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.15 Explain the Bases of each of the EOP Curves and Limits	(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	8-10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295034 G2.4.47	
	Importance Rating	4.2	

295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9: Generic K/A 2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: RO Question 26

The plant has experienced an event resulting in the following conditions:

- Reactor water level is 145 inches and rising slowly
- Reactor Building Vent Shaft radiation levels are 30 mrem/hr and rising slowly
- Reactor Water Cleanup Room temperatures are 185°F and rising slowly

The Control Room Supervisor orders DEFEAT 9, Group 3 High DW Press & RX Low Level Isolation Defeat, installation to address the conditions above.

Is this order appropriate for plant conditions and why?

A. No, DEFEAT 9 cannot be installed until RPV Level is raised above RX Lo-Lo Level Setpoint

B. No, DEFEAT 9 cannot be installed with elevated radiation levels in the reactor building vent shaft

C. Yes, DEFEAT 9 will reset the SBGT lockout relays and allow restoration of Reactor Building Ventilation

D. Yes, DEFEAT 9 will reset the SBGT lockout relays and allow SBGT to ventilate all areas of the Reactor Building

Proposed Answer: B

Explanation: EOP 3 Continuous Recheck Statement requires All the following conditions apply:
Reactor Building HVAC Isolated,
Fuel Pool Exhaust RIS4131A(B) Radiation Level is below 8 mR/hr
RB Vent Shaft RIM-7606A(B) Radiation Level is below 8 mR/hr
Offgas Vent Pipe RM-4116A(B) is belwo HI-HI Rad Trip setpoint
RB HVAC is the system normally used to maintain secondary containment temperature and Dp within operational limits. If isolated, it is appropriate to restart this system and use it to restore and maintain control of secondary containment temperature and pressure once it is confirmed that restart will not result in excessive release of radioactivity to the environment.

A. Incorrect Defeat 9 bypasses the Low RPV/L 170 inch signal and high drywell pressure it is not authorized to bypass high radiation

B. Correct: Defeat 9 does reset the L/R-5830A & B but it only bypasses the High DW (2 psig) and low RPV level (170 inch) signals. EOP-3 CRS requires fuel pool exhaust, reactor building vent shaft and offgas vent pipe radiation levels to be below their group 3 setpoints prior to installing the defeat.

C. Incorrect DEFEAT 9 cannot be installed with elevated radiation levels in the reactor building vent shaft

D. Incorrect DEFEAT 9 cannot be installed with elevated radiation levels in the reactor building vent shaft

Technical Reference(s):	EOP Defeat 9 Rev 4 EOP-3 Rev 22 Continuous Recheck Statement EOP-3 Bases Rev 13 (page 9)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.20 Relate how each step and its performance meets the mitigation strategies of the EOP support procedures	(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
	55.43/45	5/12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295036 EA2.03	
	Importance Rating	3.4	

Secondary Containment High Sump/Area Water Level - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level

Proposed Question: RO Question 27

A reactor scram occurs as a result of a seismic event. The following conditions exist:

- A break in the cooling water supply lines to the HPCI and RCIC room coolers has occurred
- RPV pressure is 900 psig and stable
- HPCI room water level is above its Max Safe Operating Level
- RCIC room water level has just risen above its Max Safe Operating Level

Which of the following describes the source of the leak and the action required to mitigate the condition?

The __(1)__ System piping is the source of the leak and an Emergency Depressurization __(2)__ required?

A. (1) ESW
(2) is

B. (1) ESW
(2) is **NOT**

C. (1) GSW
(2) is

D. (1) GSW
(2) is **NOT**

Proposed Answer: B

Explanation: Primary systems comprise the pipes, valves, and other equipment which connect directly to the RPV, a reduction in RPV pressure will effect a decrease in the flow of steam or water being discharged through an unisolated break in the system.
GSW does not supply the coolers. EOP-3 Bases point of emphasis. If a MSOL was reached in one area due to a leak and MSOL reached in another area because of a fire The BWROG guidelines were reviewed and it was concluded that ED should not be performed based on the

decisions that were previously made in step 4 (will RPV pressure reduction decrease leakage into secondary containment). An ESW leak is not a primary system leak. ED is not warranted.

A. Incorrect: ESW is the cooling water supply to the HPCI and RCIC essential room coolers.
GSW does not supply the coolers. EOP-3 Bases point of emphasis. If a MSOL was reached in one area due to a leak and MSOL reached in another area because of a fire. The BWROG guidelines were reviewed and it was concluded that ED should not be performed based on the decisions that were previously made in step 4 (will RPV pressure reduction decrease leakage into secondary containment). An ESW leak is not a primary system leak. ED is not warranted.

B. Correct
GSW does not supply the coolers. EOP-3 Bases point of emphasis. If a MSOL was reached in one area due to a leak and MSOL reached in another area because of a fire The BWROG guidelines were reviewed and it was concluded that ED should not be performed based on the decisions that were previously made in step 4 (will RPV pressure reduction decrease leakage into secondary containment). An ESW leak is not a primary system leak. ED is not warranted.

C. Incorrect
GSW does not supply the coolers. EOP-3 Bases point of emphasis. If a MSOL was reached in one area due to a leak and MSOL reached in another area because of a fire The BWROG guidelines were reviewed and it was concluded that ED should not be performed based on the decisions that were previously made in step 4 (will RPV pressure reduction decrease leakage into secondary containment). An ESW leak is not a primary system leak. ED is not warranted.

D. Incorrect
GSW does not supply the coolers. EOP-3 Bases point of emphasis. If a MSOL was reached in one area due to a leak and MSOL reached in another area because of a fire The BWROG guidelines were reviewed and it was concluded that ED should not be performed based on the decisions that were previously made in step 4 (will RPV pressure reduction decrease leakage into secondary containment). An ESW leak is not a primary system leak. ED is not warranted.

Technical Reference(s):	EOP-3 Bases Rev 13 (page 18 & 19)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	33.03.01.02 Describe the flowpath of the ESW 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	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A1.01	
	Importance Rating	4.2	

203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode: A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Reactor water level. (CFR: 41.5 / 45.5)

Proposed Question: RO Question 28

A Loss of Offsite Power has occurred and RPV Emergency Depressurization is in progress.

- “A” and “B” Core Spray pumps failed to Operate
- All remaining 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 reactor water level being to rise?

A. 450 psig

B. 330 psig

C. 260 psig

D. 135 psig

Proposed Answer: C

Explanation: EOP-1 Rev 20 Table 1A Preferred Injection Systems
Condensate/ Feedwater Shutoff Head is 650 psig (Loss of Offsite Power 1A1 and 1A2 will not transfer to the Startup Transformer on Turbine Trip and have no power)
Core Spray Shutoff Head is 330 psig Given in stem that both Core Spray systems have failed to operate.
RHR Shutoff Head is 260 psig Next available system is RHR with a shutoff head of 260 psig

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. Additionally with a LOOP 1A1 and 1A2 will be unavailable to power the condensate pumps 1P8A/B

B. Incorrect: EOP 1 identify that the nominal shutoff head for Core Spray is 330 psig. From the question stem, both core spray pumps are not available.

C. Correct: At this pressure LPCI 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 and reactor water level will rise.

D. Incorrect: At this pressure LPCI 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):	EOP-1 Table 1A Preferred Injection Systems. Rev 18 EOP-1 Bases Rev 18 (page 22 of 72)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.15 Explain the Bases of each of the EOP Curves and Limits (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
	55.45	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A2.06	
	Importance Rating	3.8	

203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode: A2.06 – Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Emergency generator failure. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 29

The plant has experienced a DBA Loss of Coolant Accident with a Loss of Offsite Power.

- Both SBDGs started and powered their respective Essential Busses
- RPV Level is stable with LPCI injection

The “B” SBDG catastrophically fails.

What is the current status of LPCI following these events?

A. A and C RHR Pumps will continue to run with no change in pump flowrate

B. A and C RHR Pumps will continue to run with elevated pump flowrates

C. A, B, C and D RHR Pumps will continue to run with no change in pump flowrate

D. A, B, C and D RHR Pumps will continue to run with elevated pump flowrates

Proposed Answer: B

Explanation: LPCI will have an initiation signal and be injecting through the select RHR Loop (B is the default injection loop). 3 RHR Pumps are required for full LPCI injection flow 14400 GPM (4800 GPM per pump). With the A & B SBDG supplying power to their respective essential buses 1A3 and 1A4 A/C will be powered by 1G31 (A SBDG), B/D will be powered by 1G21 (B SBDG). When the B SBDG fails B/D RHR pumps will trip on loss of power. A/C will remain powered but still be injecting through the full open LPCI inject valves, RHR pump flow will rise for the operating pumps.

A. Incorrect: LOCA signal will provide an auto start to the A,B,C,D RHR pumps. With the LOOP as well the RHR pumps will be powered by their respective SBDGs. When the “B” SBDG fails power will be lost to the B and D RHR pumps. With the LOCA signal RHR will select a loop for injection and open either MO-1905/2003 RHR Inboard Inject valve. The selected loop outboard inject valve has a 5 minute seal in open signal. The non-selected loop inject valves have a 10 minute seal in close signal. With a loop selected and the inject valve open with a seal in open signal the A and C pumps are running at a runout condition. Provisions are provided to allow closure / throttling of the

outboard inject valve prior to the 5 minute seal in open. Full LPCI flow is 3 pumps at rated flwo (4800 gpm/ pump or 14400 gpm)

B. Correct B and D have tripped on loss of "B" SBDG

C. Incorrect: B and D trip on loss of B SBDG

D. Incorrect only A and C RHR pumps will be running. B and D lost power with the failure of the B SBDG

Technical Reference(s):	SD-149 RHR, Rev 14 (page 9) AOP-301 rev 75 (page 12 of 58) Auto Actions Load shedding of loads	(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	(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
	55.45	6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K6.04	
	Importance Rating	3.6	

205000 (SF4 SCS) Shutdown Cooling: K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Reactor water level. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 30

The plant is in shutdown cooling when a leak developed on the running recirculating pump discharge piping.

- RPV water level is lowering

At what level will shutdown cooling **FIRST** become unavailable?

A. 190 inches

B. 170 inches

C. 119 inches

D. 64 inches

Proposed Answer: B

Explanation: Group 4 is the RHR Shutdown Cooling Isolation. Group 4 isolation signals are RPV level 170 inches, Drywell pressure 2 psig, and RPV pressure < 135 psig.
MO-1908 and 1909 SDC inboard and outboard suction valves will close on the 170 inch RPV level signal.

A. Incorrect
190” is not a SDC Group 4 signal

B. Correct
170” is the SDC Group 4 signal

C. Incorrect
119.5 ” is a Group 5 RWCU isolation signal

D. 64” is a Group 1 and 7 isolation signal

Technical Reference(s): SD 959.1 Rev 13 Primary Containment Isolation System Page 10, Table 1 (PCIS Isolation Signals) ARP 1C05A Rev 90, (Attach if not previously provided)

A-1 Reactor Lo-Lo-Lo Level Trip
B-1 Reactor Lo-Lo Level Trip
C-1 Reactor Lo Level Trip

Proposed References to be provided to applicants during examination: N

Learning Objective:	2.03.01.04 Describe the RHR system interlocks, including purpose, setpoints, logic, and when/how they are bypassed, overridden, or 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	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	7
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K4.02	
	Importance Rating	3.9	

Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCIS) design feature(s) and/or interlocks which provide for the following: System Isolation: Plant-Specific.

Proposed Question: RO Question 31

The plant is operating at 100% reactor power when a steam leak developed in the Torus Area.

- Ambient temperature has risen to 155°F and has stabilized due to Operator actions

Which of the following is correct for the current plant status?

A. RCIC will isolate in 15 minutes

B. HPCI will isolate in 15 minutes

C. HPCI and RCIC should have immediately isolated at 150°F

D. HPCI and RCIC will isolate when ambient temperature reaches 165°F

Proposed Answer: B

Explanation: At 150°F in the Torus Area a 15 minute delay starts and isolate HPCI after 15 minutes. RCIC TD is 30 minutes

A. Incorrect
RCIC isolation time delay is 30 minutes

B. Correct
HPCI isolates 15 minutes after > 150°F in Torus area

C. Incorrect
Each HPCI and RCIC will isolate after their respective time delays of 15 and 30 minutes

D. Incorrect: 165°F is the EOP-3 Max Safe Operating Limit for the Torus area but not a direct SLD input.

Technical Reference(s):
ARP 1C04B (B-4) Rev 85
SD-858 Steam Leak Detection Rev 7
SD-959.1 Primary Containment Isolation System Rev 13 (Page Page 10, Table 1 (PCIS Isolation Signals)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 5.06.01.07 Describe how the HPCI system responds to an 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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K5.01	
	Importance Rating	2.6	

209001 (SF2, SF4 LPCS) Low Pressure Core Spray: K5.01 - Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: Indications of pump cavitation. (CFR: 41.5 / 45.3)

Proposed Question: RO Question 32

The plant is experiencing a LOCA. Core Spray is injecting with the following plant conditions:

- RPV Level is +90 inches and slowly rising
- “A” Core Spray is injecting at 3300 GPM
- “B” Core Spray is injecting at 3250 GPM

The Balance of Plant Operator notices that both Core Spray Discharge pressures are oscillating.

What actions, if any, should the Operator take for Core Spray?

A. None, the Core Spray pumps are injecting within their design capabilities

B. The Core Spray pumps flow should be lowered to prevent pump runout conditions

C. The Core Spray pumps flow should be raised to meet injection requirements

D. The Core Spray pumps flow should be lowered to prevent fuel damage

Proposed Answer: B

Explanation: Section 4.0 Automatic Startup / Initiation of the Core Spray System. Step 4 verifies system parameters (flow < 3100 gpm)
Step 5 As RPV pressure lowers, throttle INBD Inject MO-2117 (MO-2137) valve using HS-2117 (HS-2137) on 1C03 to maintain <3100 gpm.
Note in section also states maintaining flow <3100 gpm is operational guidance to ensure pump run out does not occur while RPV pressure lowers

A. Incorrect:

B. Correct: Section 4.0 Automatic Startup / Initiation of the Core Spray System. Step 4 verifies system parameters (flow < 3100 gpm)
Step 5 As RPV pressure lowers, throttle INBD Inject MO-2117 (MO-2137) valve using HS-2117 (HS-2137) on 1C03 to maintain <3100 gpm.
Note in section also states maintaining flow <3100 gpm is operational guidance to ensure pump run out does not occur while RPV pressure lowers.

C. Incorrect: As RPV pressure lowers, flow will increase and should be lowered

D. Incorrect: Should be lowered to prevent pump run out Core is submerged at +15 inches

Technical Reference(s):	OI-151 Core Spray Rev 85 (page 7)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	95.00.00.14 Evaluation plant status and control room indications to determine the applicability and effect of any EOP caution	(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
	55.45	3

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A1.08	
	Importance Rating	3.3	

209001 (SF2, SF4 LPCS) Low Pressure Core Spray: A1.08 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: System lineup. (CFR: 41.5 / 45.5)

Proposed Question: RO Question 33

The plant was operating at 100% reactor power when a Recirc line break occurred. The following conditions are now present:

- Reactor pressure is at 410 psig and stable
- “A” and “B” Core Spray pumps have automatically initiated
- CORE SPRAY LOOP A INBD INJECT MO 2117 is **CLOSED**
- CORE SPRAY LOOP B INBD INJECT MO 2137 is **CLOSED**
- Core Spray MIN FLOW BYPASS VALVES MO 2104 and MO 2124 are **OPEN**

Based on the conditions above which of the following describes the response of the Core Spray System valves and if any operator actions are required?

The Core Spray Inboard Injection Valves __________(1)_______ and the Core Spray MIN Flow Bypass Valves will auto-close ________(2)________.

A. (1) should have opened and must be manually opened
(2) when Core Spray system flow reaches 600 gpm

B. (1) should have opened and must be manually opened
(2) ONLY when the Injection Valves are fully open

C. (1) will open once reactor pressure lowers to below the shut off head of the Core Spray pumps
(2) when Core Spray system flow reaches 600 gpm

D. (1) will open once reactor pressure lowers to below the shut off head of the Core Spray pumps
(2) ONLY when the Injection Valves are fully open

Proposed Answer: A

Explanation: When reactor vessel pressure drops below the low pressure permissive setpoint of 450 psig, verify that the INBD INJECT MO 2117 [MO 2137] valves OPEN to inject to the reactor vessel. The injection valves should have opened at 450 psig RPV pressure and must be opened manually based on the step stating to "verify" they open.

When system flow reaches 600 gpm, as indicated on (A[B] CORE SPRAY PUMP) INJECT/TEST FLOW indicator FI 2110 [FI 2130] on Panel 1C03, verify MIN FLOW BYPASS MO 2104 [MO 2124] valve CLOSES.

Proposed Answer: A

A. Correct - When reactor vessel pressure drops below the low pressure permissive setpoint of 450 psig, verify that the INBD INJECT MO 2117 [MO 2137] valves OPEN to inject to the reactor vessel. The injection valves should have opened at 450 psig RPV pressure and must be opened manually based on the step stating to "verify" they open.

   When system flow reaches 600 gpm, as indicated on (A[B] CORE SPRAY PUMP) INJECT/TEST FLOW indicator FI 2110 [FI 2130] on Panel 1C03, verify MIN FLOW BYPASS MO 2104 [MO 2124] valve CLOSES.

B. Incorrect - The min flow bypass valve will automatically close when system flow reaches 600 gpm.

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

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

Technical Reference(s):	OI 151, pgs 6 & 7, steps 4.0 (2) & (3). SD-151 rev 8 page 13	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	4.02.01.07 List the signals which cause a Core Spray 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	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	5
	55.45	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	21100 K3.01	
	Importance Rating	4.3	

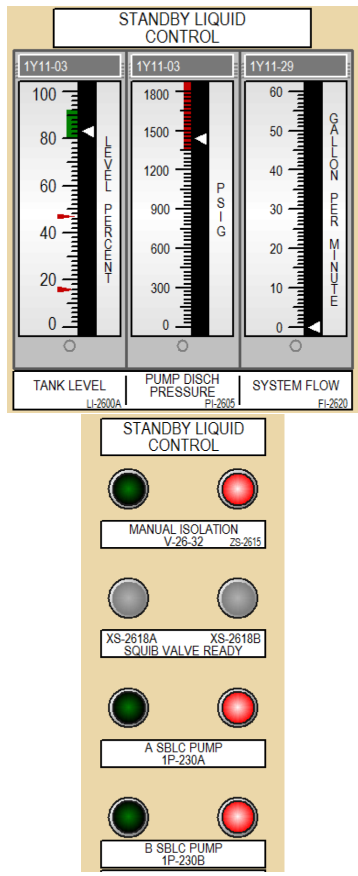
211000 (SF1 SLCS) Standby Liquid Control: K3.01 - Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: Ability to shut down the reactor in certain conditions. (CFR: 41.7 / 45.4)

Proposed Question: RO Question 34

The plant has experienced an ATWS, the Control Room Supervisor has directed the initiation of Standby Liquid Control System (SBLC).

- The SBLC system mode switch HS-2613 has been placed to “PUMPS A and B RUN”

The following is observed:



What is the status of the SBLC system?

A. The squib valves still have continuity

B. The system is operating as designed

C. The squib valves fired but the squib valves failed to provide a flow path

D. Sodium Pentaborate has precipitated out of solution blocking the pump suction line

Proposed Answer: C

Explanation:

A. Incorrect - The squib valves ready lights OFF and the squib continuity loss annunciator ON indicates that the valves have fired (lost continuity).

B. Incorrect The SLC tank level is steady and discharge pressure is above the relief valve setpoint.

C. Correct - The squib valves ready lights OFF and the squib continuity loss annunciator ON indicates that the valves have fired. However the relief valve is lifting and relieving back to the pump suction (1350 psig is the lift point) because there is no discharge path to the reactor. Of the choices given the only possible explanation is that squib valves fired but the squib valves failed to open

D. Incorrect – With the pump suction line blocked there would NOT be any discharge pressure.

Technical Reference(s):	OI 153,Rev 45 pg 6 Manual Startup/Initiation of the SBLC system OI-153 QRC Rev 4	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	6.00.00.05 Describe how the Standby Liquid Control System responds to and initiation 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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.45	4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K2.01	
	Importance Rating	3.2	

212000 (SF7 RPS) Reactor Protection: K2.01 - Knowledge of electrical power supplies to the following: RPS motor-generator sets. (CFR: 41.7)

Proposed Question: RO Question 35

Which of the following is the normal power supply to the "A" RPS MG Set and "B" RPS MG Set?

A. 1B32, CB 480VAC Essential Motor Control Center
1B42, CB 480VAC Essential Motor Control Center

B. 1B33, Turbine Building 480 VAC Motor Control Center
1B43, RB 757' Level 480 VAC Motor Control Center

C. 1B34, RB 786' Level 480 VAC Motor Control Center
1B44, RB 757' Level 480 VAC Motor Control Center

D. 1B35, RB 786' Level 480 VAC Motor Control Center
1B45, Turbine Building 480 VAC Motor Control Center

Proposed Answer: A

Explanation: 1B32 and 1B42 are the 480 VAC essential MCCs that power the RPS MG sets

A. Correct: 480 VAC Essential MCC's are the RPS MG Set power Supplies

B. Incorrect: These are 480 VAC non-essential MCC's

C. Incorrect: This is the normal supply to the 1Y11/21, Instrument AC Inverters

D. Incorrect: This is the normal supply to the 1Y23, Uninterruptible Inverter

Technical Reference(s):	SD 358 Figure #2 RPS Electrical Distribution power OI-358-A1 RPS electrical lineup Rev 3 AOP-301 Rev 75 Loss of Essential Electrical Power	(Attach if not previously provided)

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	7
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K6.01	
	Importance Rating	3.6	

212000 (SF7 RPS) Reactor Protection: K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: A.C. electrical distribution. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 36

What function does the RPS MG Set flywheel provide?

A. Minimizes the starting current on the MG

B. Maintains RPS bus voltage and frequency during momentary system faults or transients

C. Maintains the RPS bus energized for two minutes following a trip of the EPA breakers

D. Maintains the RPS bus energized during SBDG response to a loss of offsite power

Proposed Answer: B

Explanation:

A. Incorrect

B. Correct: The flywheel is designed to maintain voltage and frequency conditions during load changes and minor power interruptions

C. Incorrect

D. Incorrect

Technical Reference(s): SD-358 RPS Rev 9 (page 8) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 22.00.00.07 Describe the operation of the following principal Reactor Protection system components: (As available)

a. Reactor Protection System Motor Generator Sets

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	

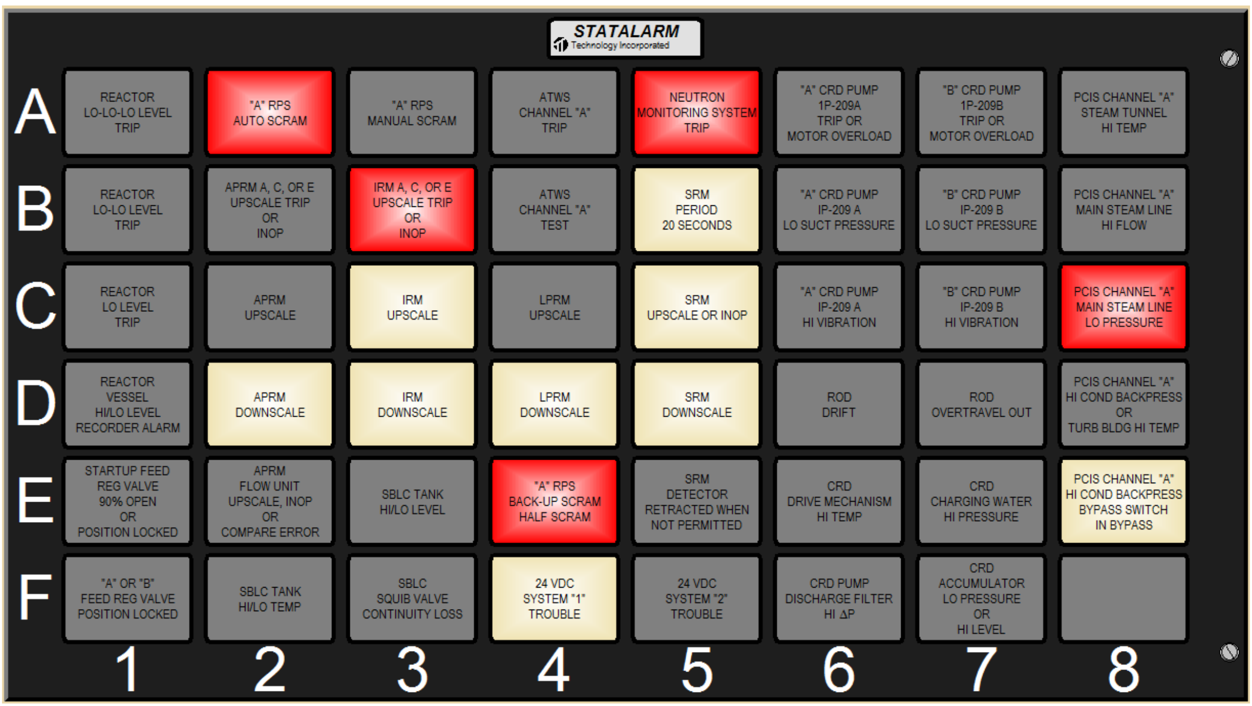
10 CFR Part 55 Content:	55.41	7
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K1.01	
	Importance Rating	3.9	

215003 (SF7 IRM) Intermediate Range Monitor: K1.01 - Knowledge of the physical connections and/or cause effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: RPS. (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Proposed Question: RO Question 37

The Mode Switch is in STARTUP when the following annunciators were received:



Which of the following caused the "A" RPS AUTO SCRAM annunciator?

A. Loss of 24 VDC System "1"

B. Main Steam Line LO Pressure

C. SRM period is less than 20 seconds

D. "A" RPS BACK-UP SCRAM valves have energized

Proposed Answer: A,

Explanation: Loss of 24 VDC with the Reactor Mode Switch not in Run will generate a RPS A (B) Half Scram

A. associated IRM levels fail low and go inoperable on the loss of 24 VDC power

B. Incorrect, MSL Low Pressure Isolation (Group 1 850 psig) is bypassed with the Mode Switch not in Run. The MSIVs will not isolate and will not cause a half or full scram signal. Though the SRMs use 24 VDC, they would fail low not high on a loss of power

C. Incorrect, This does not cause a half scram nor rod block

D. Incorrect, This is expected on an A RPS Logic Half Scram. It does not cause the ahlf scram

Technical Reference(s): AOP-375 Loss of 24 VDC Rev 27 (Attach if not previously provided)

1C05A (E-4) Rev 90

SD 358 Figure 3 RPS Trip System Simplified Rev 9 (page 42)
SD-959.1 Primary Containment Isolation System, Figure 4 Group 1 Simplified Rev 13 (page 54)

Proposed References to be provided to applicants during examination: N

Learning Objective: Given an IRM system operating mode and vatious plant conditions, p[redict how the IRM system will be impacted by failures in the following support systems:
c. DC Electrical 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 2-9

55.45 8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004	G2.4.45
	Importance Rating	4.1	

215004 (SF7 SRMS) Source Range Monitor: Generic K/A 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: RO Question 38

The reactor is in MODE 2 with a reactor startup in progress with the following conditions:

- No SRMs or IRMs are bypassed
- The SRM detectors are being withdrawn per IPOI 2

Which one of the following sets of conditions will result in activation of alarm 1C05A (E-5) SRM DETECTOR RETRACTED WHEN NOT PERMITTED

	IRM Range	SRM A (CPS)	SRM B (CPS)	SRM C (CPS)	SRM D (CPS)
A.	Range 1	120	120	120	120
B.	Range 2	90	120	120	120
C.	Range 3	90	90	90	90
D.	Range 4	90	120	120	120

Proposed Answer: B

Explanation: This alarm is bypassed when associated IRM above Range 2 or Mode Switch in RUN

A. Incorrect
SRM counts are above the level that would cause the alarm

B. Correct
SRM A is below the count value that will cause the alarm with IRMs on Range 2

C. Incorrect
With IRMs on Range 3 this alarm will not come in

D. Incorrect
With IRMs on Range 4 this alarm will not come in

Technical Reference(s): SD 878.1 Rev 7 Source Range Monitoring System Figure 10 SRM Channel Trip Outputs to RMCS page 41 (Attach if not previously provided)

ARP 1C05A (E-5) SRM Detector
Retracted when Not Permitted
Rev 90

Proposed References to be provided to applicants during examination: N

Learning Objective:	78.02.01.06 Describe the SRM system interlocks (include alarms) including the 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	10
	55.43/45	5/12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 K5.06	
	Importance Rating	2.5	

215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor: K5.06 - Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Assignment of LPRM's to specific APRM channels. (CFR: 41.5 / 45.3)

Proposed Question: RO Question 39

The plant is operating at 100% reactor power with the following plant conditions:

- “A” and “B” APRM’s are bypassed to support LPRM whisker burns
- LPRM 4D-08-09, an LPRM shared between APRM “A” and “B” fails upscale

Which of the following describes the affect of this failure on the value of computer point C179, NSSS1 CORE THERMAL POWER (MWTH)?

A. "A" APRM reading will increase causing C179 to RISE

B. "B" APRM reading will increase, however, since the APRM is bypassed C179 will REMAIN THE SAME

C. LPRMs do NOT input into the Reactor Heat Balance Equation and therefore C179 will REMAIN THE SAME

D. “B" APRM readings will lower because the "D" Level LPRM upscale reading is automatically rejected causing C179 to LOWER

Proposed Answer: C

Explanation: C179 will not be affected. LPRM inputs are not used to calculate the C179

A. Incorrect - The affect on the B APRM is correct but it has no effect on the heat balance

B. Incorrect - The affect on the B APRM is correct but it has no effect on the heat balance

C. Correct - The heat balance is used to adjust APRM gains, LPRMs and APRMs are not inputs to MWTH

D. Incorrect - LPRMs are not automatically rejected in APRMs, however in the RBM system they are.

Technical Reference(s):	SD-878.3, Rev 12; Pages 37-38 SD-900, Rev. 7, pgs. 6-8	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

SD-878.3, Rev 8; Pages 44-45.
SD-900, Rev. 4, pgs. 7-9.

Learning Objective:	80.01.01.03 Describe the interrelationships between the LPRMs, the APRMs, and the RBM systems, including the effects on one from an operation or failure of the other	(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
	55.45	3

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A4.05	
	Importance Rating	3.4	

215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor: A4.05 - Ability to manually operate and/or monitor in the control room: Trip bypasses. (CFR: 41.7 / 45.5 to 45.8)

Proposed Question: RO Question 40

The plant is at 100% reactor power with the following initial conditions:

- APRM “A” and APRM “B” are bypassed to support I&C maintenance
- An edge rod is selected

The I&C Technician inadvertently bypasses the following LPRMs associated with APRM “E”.

NOTE: The current status of the APRM “E” detectors is as shown on the chart to the right.

Prior to any Operator action, what is the expected plant response to these actions?

APRM E Detectors

LPRM	(Status)
2A-16-33	Bypassed
5A-32-17	OK
3A-16-25	OK
6A-32-09	Bypassed
3B-24-25	OK
4B-08-09	Bypassed
4B-08-17	Bypassed
5B-40-17	OK
3B-24-33	OK
3C-32-33	Bypassed
4C-16-17	OK
1C-16-41	Bypassed
4C-32-25	OK
5C-16-09	OK
1D-24-41	Bypassed
2D-08-25	Bypassed
3D-40-25	OK
4D-24-09	OK
2D-08-33	OK
4D-24-17	OK

A. RBM Downscale

B. Control Rod Block, ONLY

C. Half SCRAM and Control Rod Block

D. Full SCRAM and Control Rod Block

Proposed Answer: C

Explanation: APRM E requires 13 LPRM inputs to be Operable. In current configuration it only has 12 LPRM inputs. This would generate an INOP signal from “E” APRM

A. Incorrect, no re-null signal, provided in the stem, an edge rod is selected. RBM is bypassed so, no RBM downscale will be generated. This would be possible provided a non-edge rod was selected.

B. Incorrect, APRM E is not bypassed, so an A RPS INOP will also be generated

C. Correct, too few LPRM inputs will generate the E APRM INOP signal causing a Rod Block and A Side Half Scram

D. Incorrect, not a full scram signal
too few LPRM inputs will generate the E APRM INOP signal causing a Rod Block and A Side Half Scram

Technical Reference(s):	ARP 1C05A (B-2) SD 878.3 Rev 12 page 37, 38 OI-878.4 Rev 43, P&L 3	(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 system or components: c. LPRMS	(As available)

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

Question History: Last NRC Exam: 2013

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

10 CFR Part 55 Content:	55.41	7
	55.45	5-8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A3.02	
	Importance Rating	3.6	

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling: A3.02 - Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: Turbine startup. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 41

The RCIC Turbine is in service aligned to its normal suction path for RPV injection when the following annunciator is received:

- 1C04C D-5 RCIC LOW SUCTION PRESSURE

Which one of the following describes the automatic operation of the RCIC turbine following this condition?

A. RCIC suction will swap from the CST to the TORUS and continue to operate

B. RCIC suction will swap from the TORUS to the CST and continue to operate

C. RCIC will AUTO ISOLATE

D. RCIC Turbine will TRIP

Proposed Answer: D

Explanation: Low suction pressure is a RCIC Turbine Trip

A. Incorrect, suction swap from CST to Torus is on CST low level, not low suction pressure

B. Incorrect, no low suction pressure suction swap

C. Incorrect, this is not an Auto Isolation signal

D. Correct

Technical Reference(s):
ARP 1C04C (D-5)
SD-150 Rev 9 Page 10-11
OI-150 Rev 88 Page 50 (Appendix 1 RCIC Turbine Trips and Isolations)
(Attach if not previously provided)

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	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	7
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 A2.02	
	Importance Rating	3.5	

218000 (SF3 ADS) Automatic Depressurization: A2.02 - Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Large break LOCA. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 42

The ADS function is based on the ___(1)____ system being unable to maintain reactor water level during a LOCA and during EOP execution the ADS System ___(2)___ bypassed to prevent automatic operation.

A. (1) HPCI
(2) is

B. (1) HPCI
(2) is **NOT**

C. (1) LPCI
(2) is

D. (1) LPCI
(2) is **NOT**

Proposed Answer: A

Explanation: The purpose of the ADS system is to provide an automatic means of reducing reactor pressure for events such as pipe breaks or reactor loss of water level transients when the HPCI system is unable to maintain reactor water level. The pressure reduction enables low pressure injection systems such as LPCI and Core Spray to inject additional makeup water into the vessel to restore or maintain water level preventing overheating of the fuel cladding.
ATWS ADS is locked out to prevent uncontrolled injection of large amounts of relatively cold unborated water form low pressure systems. Such an occurrence would quickly dilute in-core boron concentration and reduce reactor coolant temperature. When the reactor is not shutdown, or when the shutdown margin is small, sufficient positive reactivity might be added in this way to cause a reactor power excursion large enough to severely damage the core.
EOP-1 CRS if ADS Timer has initiated Lockout ADS: Undesirable for the following reasons: ADS actuation can cause impose a severe thermal transient on RPV and may complicate efforts to control RPV level

If only steam driven systems are available for injection ADS actuation may directly lead to loss of adequate core cooling.
Crew can draw on much more information than is available to ADS logic and can better judge based on instructions contained in the EPGs/SAGs when and how to depressurize the RPV

A. Correct purpose of ADS, ADS is overridden per EOP directions

B. Correct purpose for ADS, however ADS is overridden per EOP directions

C. Incorrect, ADS lowers pressure so LPCI can inject, ADS is overridden per EOP directions

D. Incorrect, ADS lowers pressure so LPCI can inject, ADS is overridden per EOP directions

Technical Reference(s):	SD-183 Rev 9, ALC leg of EOP-1 RC/L-2 locks out ADS Rev 20 EOP-1 Continuous Recheck Statement ATWS /-1 Locks Out ADS Rev 23	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	8.00.00.01 State the purpose of the ADS 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	5
	55.45	6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A1.02	
	Importance Rating	3.7	

223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff: A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Valve closures. (CFR: 41.5 / 45.5)

Proposed Question: RO Question 43

Which of the following conditions will **DIRECTLY** cause a Primary Containment Isolation to occur?

A. RPV Level at 211 inches

B. RPV Pressure at 1140 psig

C. HPCI Steam Line Flow at 150%

D. RWCU HX Room ambient temperature at 135°F

Proposed Answer: D

Explanation:

A. Incorrect, RPV level at 211” is a RFP and Turbine trip but not a direct PCIS signal

B. Incorrect, 1140 PSIG is RPT breaker trip signal, not a direct PCIS signal

C. Incorrect, HPCI PCIS isolation is 300%

D. Correct: PCIS Group 5 signal is 130°Fin the RWCU Hx Room

Technical Reference(s): SD-959.1 Rev 13 (page 10) Table 1 PCISW Isolation Signals (Attach if not previously provided)

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	5
	55.45	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K6.04	
	Importance Rating	3.0	

239002 (SF3 SRV) Safety Relief Valves: K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: D.C. power: Plant-Specific. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 44

The plant is operating at 100% reactor power when the following events occur:

- 1D13 Circuit 14 "AUTO BLOWDOWN RELAY PANEL 1C45" trips
- Annunciator 1C03A (C-6) ADS/LLS 125 VDC CONTROL POWER FAILURE alarms

Regarding ADS operation, which ONE of the following describes the effect of the breaker trip?

A. "A" ADS logic shifts to its alternate power supply so there is temporary loss of power to the ADS logic

B. "A" ADS logic has lost power; however, all 4 ADS SRVs have control power, and there is NO effect on the operation of ADS

C. "A" ADS logic has lost power; however, ONLY PSV 4401 and PSV 4407 have alternate control power and will open during ADS initiation

D. "A" ADS logic shifts to its alternate power supply; however, control power is lost to PSV-4402 and PSV-4405, therefore, these valves will NOT open during ADS initiation

Proposed Answer: B

Explanation: ADS Logic “A” does not have a backup 125 VDC supply

A. Incorrect, “A” ADS Logic has no backup 125 VDC Power supply

B. Correct
ADS Relief Valves PSV4400,4402, 4405, 4406 have control power via 1D23-14

C. Incorrect, all ADS PSVs will function. 4401 and 4407 are LLS SRVs

D. Incorrect, A ADS Logic has no backup. All ADS SRVs will shift to backup power 1D23-14

Technical Reference(s): SD-183 (Attach if not previously provided)

ADS and LLS DC Power Supplies
page 22 (rev 9)

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 system will be impacted by failures in the following support systems: d. 125 VDC buses 1D1 and 1D2	(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
	55.45	7

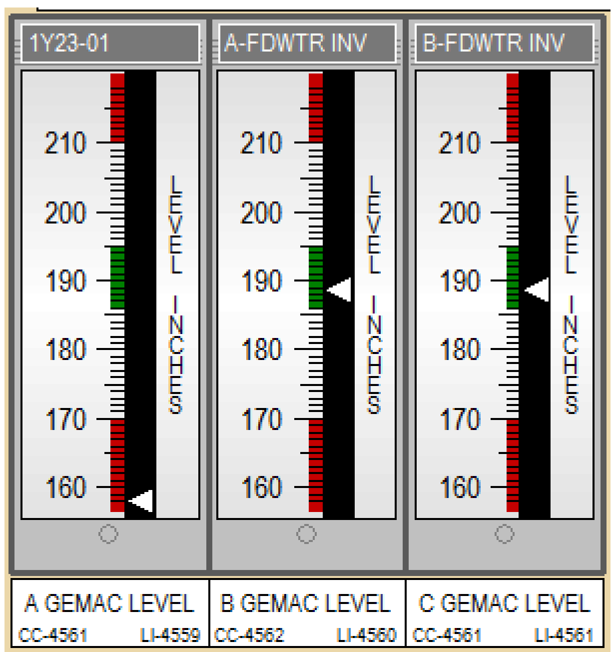
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K5.03	
	Importance Rating	3.1	

259002 (SF2 RWLCS) Reactor Water Level Control: K5.03 - Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: Water level measurement. (CFR: 41.5 / 45.3)

Proposed Question: RO Question 45

The plant is operating at 100% reactor power.

The Operator at the Controls observes the following:



What action, if any, should the Operator take?

A. SCRAM the Reactor

B. No Operator actions are required

C. Place HS 4450 1 OR 3 ELEMENT CONTROL SELECT to 1 ELEMENT

D. Place HS 4560 RX WATER LEVEL CONTROL INPUT SELECTED to C-LEVEL

Proposed Answer: B

Explanation: OI-644 P&L 14
Due to A/C GEMACS sharing a common variable leg, the “B” level control instrument is preferred. RX WATER LEVEL CONTROL INPUT SELECT HSS-4560 should normally be selected to B Level.
AOP 302.1 Loss of 1D11 with “B” Level Selected and no operator actions results in
Feedwater Control opens Feed Reg Valves
RPV level goes high
“B” RFP and Main Turbine Trip on high RPV level (A RFP cannot be tripped remotely)
Reactor Scram (turbine trip)

A. Incorrect
RPV Level is in the normal band. A Gemac has failed downscale, no scram criteria has been met.

B. Correct see explanation above…

C. Incorrect
B Level input is functioning correctly and no need to select 1 element

D. Incorrect
HSS4560 has two positions A Level and B Level. B is normally selected and there is no criteria met to transfer to A Level control.

Technical Reference(s):	OI-644 Rev 182 P&L 14 page 7 AOP-644 Rev 23 (page 3 step 5) SD 644 Rev 5, Figure 5 Feedwater Control System AOP-302.1 Loss of 125 VDC Rev 59 1D11 Auto Actions page 2 (1D11 was not lost) no action required	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	45.05.01.05 Describe the operation of the feedwater control circuitry include: a. RPV level control circuitry	(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
	55.45	3

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K4.05	
	Importance Rating	2.6	

261000 (SF9 SGTS) Standby Gas Treatment: K4.05 - Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Fission product gas removal. (CFR: 41.7)

Proposed Question: RO Question 46

What design feature of the Standby Gas Treatment System provides for the removal of Fission Product Gases?

A. Prefilter

B. HEPA Filter

C. Roughing Filter

D. Carbon Bed Filter

Proposed Answer: D

Explanation:

A. Incorrect: removes particulates

B. Incorrect: Remove particulates greater than 0.3 micron in size

C. Incorrect: removes moisture

D. Correct
The activated carbon iodine filter is a high efficiency deep bed type with a 6 inch layer of charcoal. Each train contains approx. 1224 pounds of potassium iodide impregnated activated charcoal for trapping elemental iodine and radioiodine in the form if organic compounds.

SD 170 Rev 14 (page 7)

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	7.04.01.06 describe the operation of the following principle SBGT system components: d. Charcoal Adsorber	(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	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K3.03	
	Importance Rating	2.9	

262001 (SF6 AC) AC Electrical Distribution: K3.03 - Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: D.C. electrical distribution. (CFR: 41.7 / 45.4)

Proposed Question: RO Question 47

A lockout occurs on 4160 VAC Bus 1A3, what is the expected response to the 125 VDC distribution system?

A. The 1D12, Division 1 Battery Charger will automatically align to 1A4

B. The 1D120, 125 VDC Swing Charger, will automatically align to 1D10

C. The Division 1, 125 VDC system will be powered by station battery

D. The Division 1, 125 VDC system will be de-energized

Proposed Answer: C

Explanation: The battery chargers normally supply system power with the batteries providing supplemental power during periods of high demand. The batteries provide at least 4 hours of complete backup power during battery charger failure. The 125 VDC system is arranged into two redundant and separate subsystems each consisting of a battery, battery charger, distribution apparatus and the necessary instrumentation, controls, and protective devices required to satisfy the system functional and design objectives. A third or swing charger is provided as a permanently connected spare for either of the normal battery chargers, This third charger can also be fed from either essential AC bus via interlock.
The third or swing battery charger (1D120) is connected via a pair of interlocked breakers to either 125 VDC panels. AC input to 1D120 is fed from 1B32x42 which interlocks AC input from either 1B3202 or 1B4210A. Only one battery charger for each of the 125 VDC panels is in service , while the third charger (1D120) is a spare for either panel. A mechanical interlock on this charger prevents it from supplying both 125 VDC panels.

A. Incorrect: There is no auto power swap for the station battery chargers

B. Incorrect: 1D120 the swing charger does not automatically realign

C. Correct

D. Incorrect,:the station will carry loads up to 4 hours

Technical Reference(s): SD-375 Rev 10 (page 4, 8, 13) (Attach if not previously provided)

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 #		
	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
	55.45	4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 K1.19	
	Importance Rating	2.9	

262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC): K1.19 - Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Power range neutron monitoring system: Plant-Specific. (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Proposed Question: RO Question 48

What is the impact to the APRM/LPRM system if a loss of 1Y23, 120V Uninterruptible Power Supply occurs?

The ____(1)____ will immediately ____(2)____.

A. (1) NMR9254A, NMR9254C, IRM/APRM recorders
(2) de-energize

B. (1) NMR9254A, NMR9254C, IRM/APRM recorders
(2) indicate downscale

C. (1) NMR9254B, NMR9254D, IRM/APRM recorders
(2) de-energize

D. (1) NMR9254B, NMR9254D, IRM/APRM recorders
(2) indicate downscale

Proposed Answer: C

Explanation: AOP 357 probable indications: NMR-09254B/D screen goes dark
AOP-317 Loss of Inst AC Probable indications NMR-9254A/C oss or failure

A. Incorrect: Inst AC power not uninterruptible
NMR-9254B/D screens go blank

B. Incorrect: Inst AC power not uninterruptible
NMR-9254B/D screens go blank

C. Correct: AOP-357 NMR-9254B/D screens go blank

D. Incorrect NMR-9254B/D screens go blank

Technical Reference(s): AOP-357(rev 50) (Attach if not previously provided)

AOP-317 Rev 109 (page 10)

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: g. Instrument AC Power	(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	2-7
	55.45	7-8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K.2.01	
	Importance Rating	3.1	

263000 (SF6 DC) DC Electrical Distribution: K2.01 - Knowledge of electrical power supplies to the following: Major D.C. loads. (CFR: 41.7)

Proposed Question: RO Question 49

With the plant operating at 100% reactor power.

Which of the following describes the impact to the loss of 1D40, 250V DC Distribution Panel?

A. HPCI cannot operate as designed

B. RCIC cannot operate as designed

C. Reactor Recirc MG Sets will trip

D. The Main Turbine will trip

Proposed Answer: A

Explanation: HPCI valves and pumps are 250 VDC power (exception MO2238 inboard steam supply). Without 250 VDC the Aux Oil pump will not start to provide the initial oil pressure required to initiate HPCI, MO2312 normally closed discharge valve has no power to open and allow HPCI to inject.

A. Correct

B. RCIC is 125 VDC power

C. Incorrect. The RRMG Set Emergency Lube Oil Pumps will lose power but this will not cause a loss of lube oil pressure trip

D. Incorrect. The EBOP is 250 VDC, but loss will not cause a TT

Technical Reference(s): AOP-388 Rev 21 page 7) 1D41 load list (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	5.01.01.02 Given a HPCI system operating mode and various plant conditions predict how the HPCI system will be impacted bu the following support system failures: c. 250 VDC Distribution	(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	

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	264000	A4.05	
	Importance Rating	3.6		

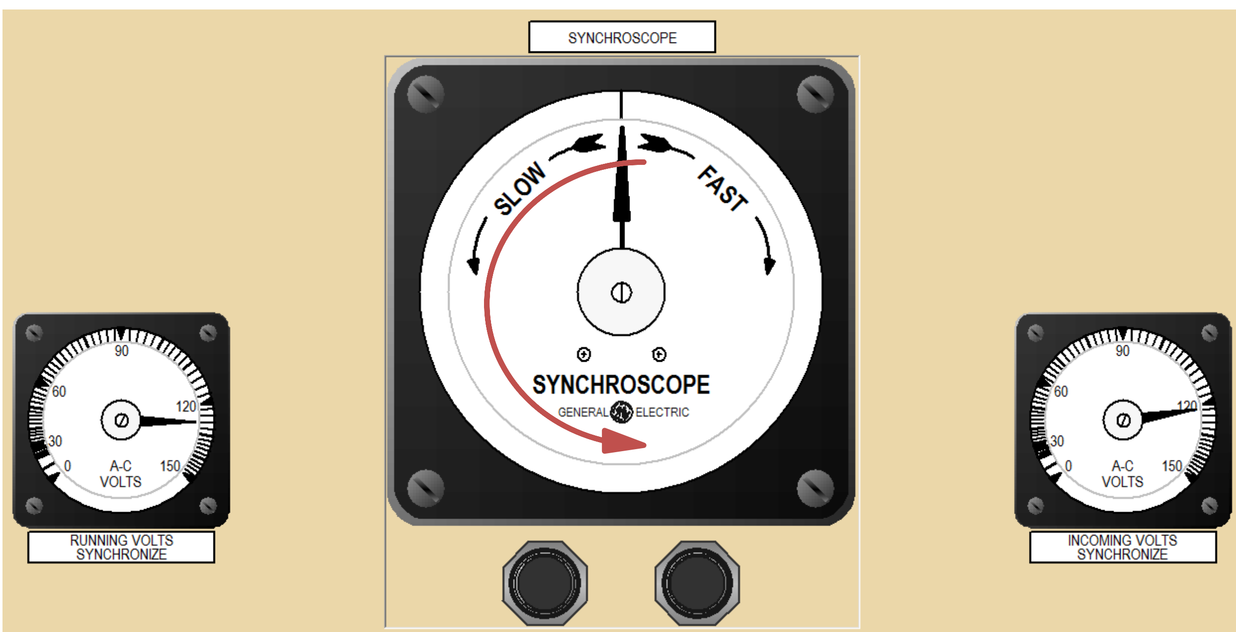
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG: A4.05 - Ability to manually operate and / or monitor in the control room: Transfer of emergency generator (with load) to grid. (CFR: 41.7 / 45.5 to 45.8)

Proposed Question: RO Question 50

The Standby Diesel Generator (SBDG) is being paralleled to the grid in accordance with OI 324, Standby Diesel Generator System.

The following indications are observed:

- The synchroscope is rotating at 3 RPM
- The synchroscope is rotating in the **SLOW** direction (counter clockwise)



Which of the following is the correct action to parallel the SBDG to the grid?

The Operator ______.

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

B. cannot close the SBDG 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 SBDG to the grid

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

Proposed Answer: D

Explanation: OI324 it is required to have a slow clockwise rotation of sync scope and incoming voltage slightly higher than running Section 6.5 and 6.6

A. Incorrect: per OI it is required to have a slow clockwise rotation of sync scope and incoming voltage slightly higher than running

B. Incorrect: normal indication with scope needle at 12 position

C. Incorrect: per OI it is required to have a slow clockwise rotation of sync scope and incoming voltage slightly higher than running

D. Correct: per OI it is required to have a slow clockwise rotation of sync scope and incoming voltage slightly higher than running

Technical Reference(s):	OI-324 rev 125 page section 6.5 and 6.6 Paralleling to 1A3 / 1A4 page 34, 38)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	19.01.01.03 Describe the operation of the following SBDG components and controls: g. governor 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	7
	55.45	5-8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.03	
	Importance Rating	2.8	

300000 (SF8 IA) Instrument Air: K4.03 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water. (CFR: 41.7)

Proposed Question: RO Question 51

- 1K-1, Backup Instrument Air Compressor, power is aligned to 1B45
- 1K-1 is the only available air compressor operating to support plant air needs
- “A” and “C” Well Water Pumps are operating

A loss of 1B3 occurs.

With no Operator action, 1K-1 will _____________.

A. trip on high first stage air outlet temperature

B. continue to be cooled by GSW

C. trip due to the loss of motor cooling

D. continue to be cooled by Well Water

Proposed Answer: A

Explanation: 1B33 supplies power to 1P58A and C Well pumps which are lost on 1B3 loss
GSW is the backup cooling to 1K1 and requires manual re-alignment
Motor is air cooled

A. Correct, loss of well water cooling will cause a high air outlet trip

B. Incorrect: On loss of 1B3 MCC 1B33 will be lost thereby causing the loss of 1P-58A and C well water pumps. GSW is the backup cooling water, but requires manual re-alignment

C. Incorrect: 1K1 motor is air cooled

D. Incorrect: On loss of 1B3 MCC 1B33 will be lost thereby causing the loss of 1P-58A and C well water pumps. GSW is the backup cooling water, but requires manual re-alignment

Technical Reference(s):	SD-518 Section 7.9 Shifting 1K-1 Cooling Water from Well Water to GSW (page 58) Rev 103 AOP-301 rev 75 (page 47 1B33 loads 1P58A/C). SD-518 rev 9, page 13	(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 system in the correct condition	(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	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 G2.1.28	
	Importance Rating	4.1	

300000 (SF8 IA) Instrument Air: A3.02 - G2.1.28 - Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

Proposed Question: RO Question 52

Which of the following is a plant function directly supported by the 1K-3, CB/SBGT Instrument Air Compressor?

A. Breathing air for the control room

B. Outboard Main Steam Line Isolation Valves

C. Secondary Containment Isolation Dampers

D. Primary Containment Isolation Damper/Valve T-Seals

Proposed Answer: D

Explanation: 1K-3/4 support primary containment vent and purge valve T-seals.
SD 959 Primary Containment Control valves are equipped with inflatable T-ring seals that provide a leak-tight seating surface for butterfly discs on the valves.
SD-573 Containment Purge and Vent Valves the 18 inch containment purge and vent valves are pneumatically operated, butterfly dampers with a fail close actuator. An isolation signal causes the air supply solenoid to de-energize which vents air from the actuator and allows actuator spring force to close the valve. As the valve closes a mechanical arm will trip a limit switch, which pressurizes T-ring seals. These T-ring seals provide a seal around the valve disc to prevent any leakage from the valve disc and seat area.

A. Incorrect: 1K-3/4 do not supply breathing air. This is via the main plant air system

B. Incorrect: N2 is the motive force for the outboard MSIVs

C. Incorrect: Main plant air supplies secondary containment isolation dampers

D. Correct

Technical Reference(s):	SD-573 rev 15 page 9 OI-730 rev 126 table 1 1K-3 and 1K-4 Technical Specification Related loads (page 65). SD-959 rev 6 page 10	(Attach if not previously provided)

TS Bases 3.6.1.3 A.1 and A.2
page 3.6-19.
TS Bases 3.7.9 Background page
B3.7-40 (TSCR-044A)

Proposed References to be provided to applicants during examination: N

Learning Objective:	7.00.00.02 given a SBGT system operating mode and various plant conditions predict how the SBGT system will be impacted by failures in the following support systems: b. Instrument and Service air	(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	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A3.01	
	Importance Rating	3.0	

400000 (SF8 CCS) Component Cooling Water: A3.01 - Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS (CFR: 41.7 / 45.7)

Proposed Question: RO Question 53

A leak in the RBCCW system from a NRHX tube leak has occurred, how will the level in the surge tank respond?

A. Lower, automatic makeup to refill the surge tank

B. Raise, the expansion tank will overflow to the floor drain system

C. Lower, manual operator action is required to refill the surge tank

D. Raise, manual operator action is required to vent the surge tank

Proposed Answer: B

1T-78 overflows to open radwaste in the case of high water level. Any leakage from components at reactor pressure will be into the RBCCW system. Since it is desired to determine leakage into or out of the RBCCW system, makeup water is normally manually isolated and water level is maintained by manually adding demineralized water to the tank.

Explanation:

A. Incorrect, level will rise and auto makeup is isolated

B. Correct

C. Incorrect, level will rise and auto makeup is isolated

D. Incorrect, level will rise and auto makeup is isolated. Surge tank is vented to atmosphere

Technical Reference(s): SD 414 rev 9, page 8 & 9
OI-414 rev 42, P&L 1 page 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	29.03.01.02 for any given RBCCW system operation or failure, describe the impact of that operation or failure on the following systems or components: g. RWCU Non-regenerative HX	(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
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 K3.03	
	Importance Rating	3.1	

201001 (SF1 CRDH) CRD Hydraulic: K3.03 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following: Control rod drive mechanisms. (CFR: 41.7 / 45.4)

Proposed Question: RO Question 54

The plant is operating at 100% reactor power when a loss of air to the in service CRD Flow Control Valve occurs.

Which one of the following describes the impact on the CRD System?

A. Control Rods may drift

B. Control Rod scram times will lower

C. CRD Accumulators would begin to discharge

D. CRD Mechanism seal degradation would accelerate

Proposed Answer: D

Explanation: Flow Control Valves CV-1821 and 1822 fail closed (AOP-518) With CV-1821 / 1822 closed cooling flow to the CRD drives is decreased and drive temperatures rise. 1C05A (E-6) CRD Drive Mechanism HI Temp alarms at 250°F. Although the drives can function without cooling water, seal and bushing life is shortened by long term exposure to reactor temperatures.

A. Incorrect – the flow control valve fails closed on a loss of air reducing the pressure on the underside of the CRDM. This is plausible in the if the valve failed open it would increase the pressure on the underside of the CRDM piston resulting in the potential for rods drifting in.

B. Incorrect - The flow control valve fails shut on loss of air however charging water pressure is unaffected.

C. Incorrect – The charging line is upstream of the FCV

D. Correct – the FCV fails shut on loss of air. Cooling water flow to the drives is reduced causing increased CRDM temperatures. Continued operation with elevated CRDM temperatures could result in CRDM seal degradation.

Technical Reference(s): AOP-518 Rev 42 page 2 (Automatic Actions) SD 255 Rev 9 (page 15, & 23) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	10.01.01.05 given a CRDH system operating mode and various plant conditions, predict how the CRDH system will be impacted by the failure of the following support systems: f. Instrument and service air	(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

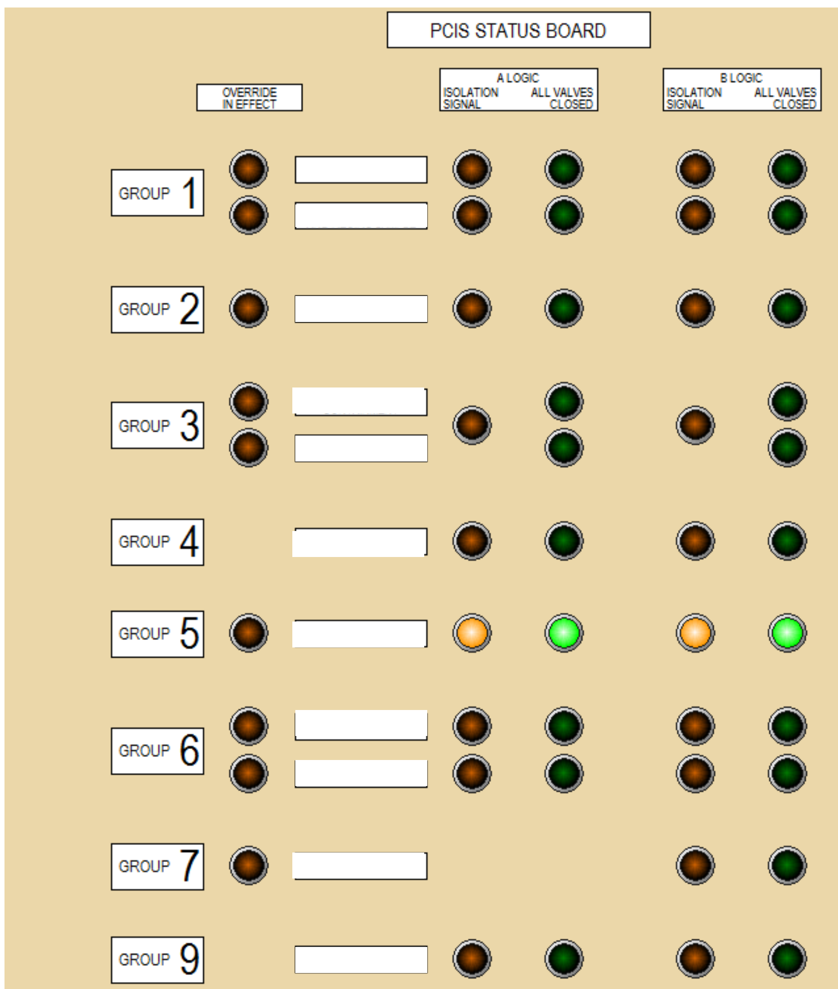
10 CFR Part 55 Content:	55.41	7
	55.45	4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K1.11	
	Importance Rating	3.5	

204000 (SF2 RWCU) Reactor Water Cleanup: K1.11 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: PCIS/NSSSS. (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Proposed Question: RO Question 55

The PCIS isolation demonstrated by the CIMS panel affected which system?



A. TIP

B. RHR

C. RWCU

D. Main Steam

Proposed Answer: C

Explanation:SD 959.1 Primary Containment Isolation System Group 5 affects the RWCU system.

A. Incorrect this would be Group 2

B. Incorrect, this would be Group 4

C. Correct this is affected by the Group 5 isolation

D. Incorrect this would be Group 1

Technical Reference(s):	SD 959.1 Rev 13 page 10 Table 1 PCIS Isolation Signals PCIS Status Board picture in SD 959.1 page 35 (rev 13)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	50007.05.05 Describe the indications and meaning of the signals provided by the CIMS panel	(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	2-9
	55.45	7-8

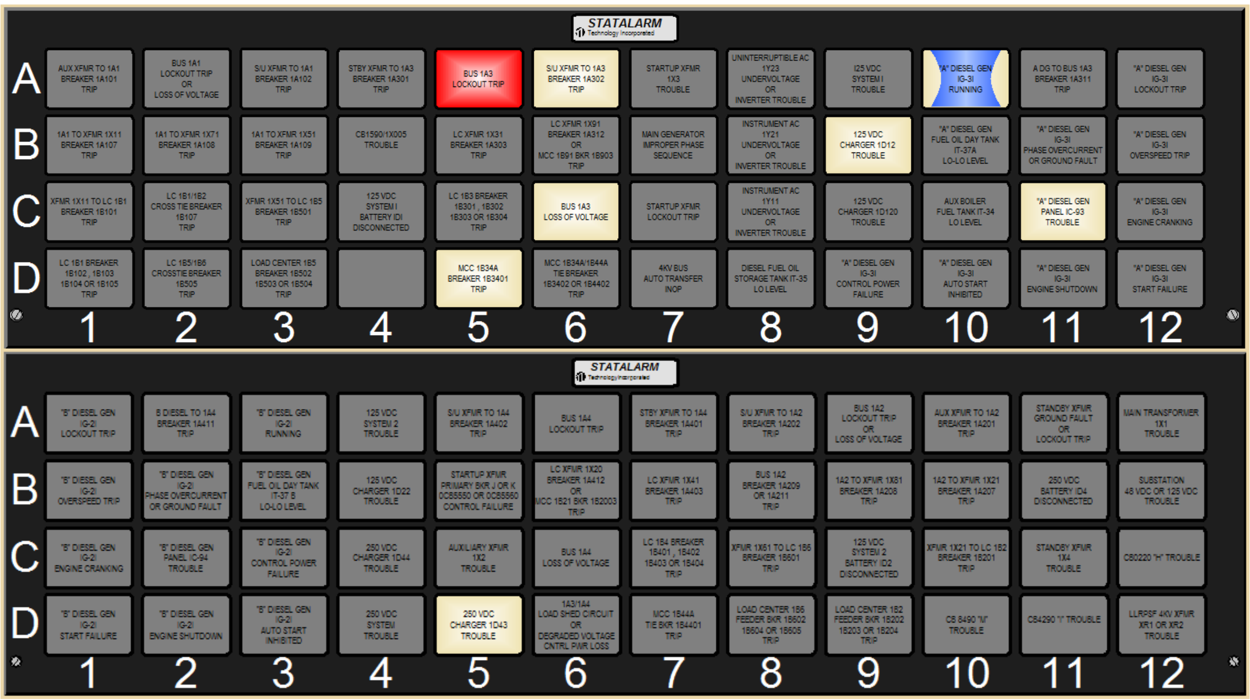
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	219000 K2.02	
	Importance Rating	3.1*	

RHR/LPCI: Torus/Suppression Pool Cooling Mode: Knowledge of electrical power supplies to the following: Pumps. (CFR: 41.7)

Proposed Question: RO Question 56

The RHR System is in torus cooling mode with “A” RHR Pump and “B” RHR Pump operating.

The following is observed at 1C08:



What is the status of the “A” RHR and “B” RHR pumps based upon the following indications?

A. “A” RHR pump is operating
“B” RHR pump is operating

B. “A” RHR pump is operating
“B” RHR pump is NOT operating

C. “A” RHR pump is NOT operating
“B” RHR pump is operating

D. “A” RHR pump is NOT operating
“B” RHR pump is NOT operating

Proposed Answer: C

Explanation: Bus 1A3 Lockout Trip Automatic Actions: Breakers !a301 and 1A302 trip open and are interlocked from manually or automatically closing AND Bus 1A3 Load Sheds; A SBDG 1G31 auto starts on Bus 1A3 undervoltage, then runs up to speed and frequency. Breaker 1A311 does not auto close and cannot be manually closed.
AOP 301 Load Shedding of the following loads: 1P-229A and C RHR Pumps.

A. Incorrect: 1A3 lockout will cause a loss of voltage to 1A3, prevent 1A311 SBDG output breaker from closing onto the bus. 1P229A “A” RHR pump will not be running

B. Incorrect: 1P229B would still be running 1A4 still has power and no trip conditions are given.
1A3 lockout will cause a loss of voltage to 1A3, prevent 1A311 SBDG output breaker from closing onto the bus. 1P229A “A” RHR pump will not be running

C. Correct

D. Incorrect: 1P229B would be running, no trip signal is present for it.
1A3 lockout will cause a loss of voltage to 1A3, prevent 1A311 SBDG output breaker from closing onto the bus. 1P229A “A” RHR pump will not be running

Technical Reference(s):	ARP 1C08A (A-5) rev 100 page 18 AOP 301 Loss of 1A3 rev 75 (page 2) Load Shedding information	(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 #		(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	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	233000	G2.4.18
	Importance Rating	3.3	

233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup: Generic K/A 2.4.18 - Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)

Proposed Question: RO Question 57

What is the basis for the EOP entry condition for Fuel Pool Temperature?

To prevent ______.

A. exceeding the K_{eff} of the spent fuel in the pool

B. exceeding the design limitation of the fuel pool and system components

C. exceeding the Environmental Qualification of the fuel pool level indicators

D. the loss of Net Positive Suction Head to the Spent Fuel Pool Cooling pumps

Proposed Answer: B

Explanation: UFSAR 9.1.2.3.2.2 the limiting design factor of the FPCCU System is the piping designed to accommodate 150°F water at 200psig. Spent fuel pool design temperature is 150°F The spent fuel pool design temperature is based on the capability of the FPC system piping , which bounds the temperature limitations of the pool liner

A. Incorrect: Fuel pool temperatures at the RPC pump suction shall be kept greater than 40°F when the reactor cavity to fuel pool gates are installed and 68°F during refueling operations when the gates are removed. UFSAR.

B. Correct

C. Incorrect:

D. Incorrect SFPC pumps trip on low skimmer surge tank level. Per EOP 3 the minimum safe operating spent fuel pool level is 25.17 ft (10 ft above the top of the fuel racks) well below the SST low level trips.

Technical Reference(s): EOP-3 bases rev 13, (pages 24-25)
UFSAR 9.1
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: 95.00.00.15 Explain the Bases of each of the EOP Curves and Limits (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

55.43/45 1/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	239001 A4.11	
	Importance Rating	3.1	

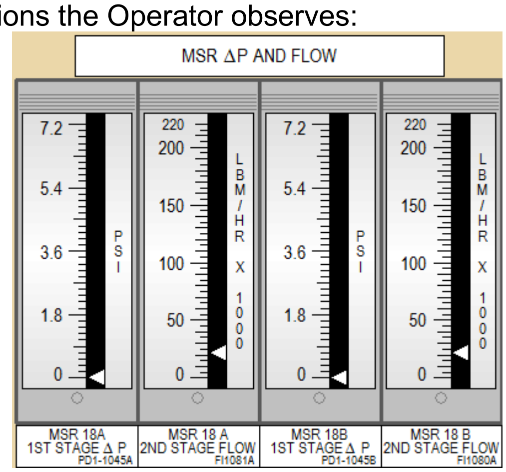
239001 (SF3, SF4 MRSS) Main and Reheat Steam: A4.11 - Ability to manually operate and/or monitor in the control room: Alternate methods of verifying valve positions. (CFR: 41.7 / 45.5 to 45.8)

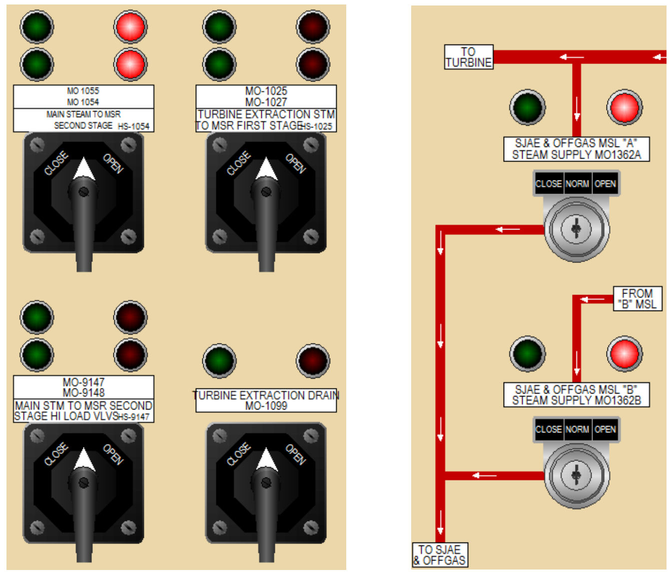
Proposed Question: RO Question 58

The plant was operating at power with the Startup Transformer removed from service.

- The crew inserted a manual reactor scram
- The main turbine has tripped
- Reactor pressure is 940 psig and stable

While performing IPOI 5 actions the Operator observes:





What, if any, actions should the Operator perform in accordance with IPOI 5, Scram?

A. Close MO-1054 and MO-1055

B. Close MO-1362A and MO-1362B

C. Close MO-9147 and MO-9148

D. No action is required for the observed condition

Proposed Answer: A

Explanation: Turbine Trip with the Startup Transformer out of service will cause a loss on nonessential electrical power.
MO9147 is powered by 1B1234 (non-essential)
MO9148 is powered by 1B2224 (non-essential)
MO-1362A is powered by 1B3705 (essential power)
MO1362B is powered by 1B3706 (essential power)
MO1054 is powered by 1B3708 (essential power)
MO-1055 is powered by 1B3707 (essential power)
IPOI-5 QRC step 9 is monitor RPV pressure, maintain RPV pressure below 1110 psig.

- Verify close MO-9147 / 9148 Close MO-1054 / 1055 if necessary

IPOI-5 If steam flow is still indicated on either FI-1080A or FI-1081A, then close MO-1054 / 1055 MN STM to MSR Second Stage

A. Correct: as directed by IPOI-5 and QRC. Flow is still indicated on FI's after scram. On loss of non-essential power caused by the turbine trip MO-9147 and 9148 will have lost power. It is necessary to close MO-1054 and 1055

B. Incorrect

C. Incorrect: Loss of non-essential power will prevent these valves from closing or being closed

D. As seen by picture steam flow is present requiring MO-1054 and 1055 to be clsoed

Technical Reference(s):	IPOI-5 rev 62 (page 6 of 15) IPOI-5 QRC rev 12 (page 1 of 1)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	93.00.00.14 Contrast the different methods of cooling down the reactor when the Main Condenser is and is not available	(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
	55.45	5-8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000 K5.04	
	Importance Rating	3.3	

241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating: K5.04 – Knowledge of the operational implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM: Turbine inlet pressure vs. reactor pressure. (CFR: 41.5 / 45.3)

Proposed Question: RO Question 59

The plant is operating at 98% reactor power with the "A" EHC Pressure Regulator in service.

- The "A" Pressure Regulator malfunctions such that "Steam Throttle Pressure A" is slowly failing DOWNSCALE

Assuming NO operator action is taken, which of the following correctly describes the expected response of turbine throttle pressure?

Turbine throttle pressure will __________.

A. slowly rise until the reactor scrams on either high flux or high pressure

B. stabilize a few psig lower controlled by the "B" EHC Pressure Regulator

C. slowly lower resulting in a reactor scram on an automatic MSIV closure

D. stabilize a few psig higher controlled by the "B" EHC Pressure Regulator

Proposed Answer: D

Explanation: As the “A” steam throttle pressure falls, the “A” side pressure error signal will go down. This will cause the CVs to begin to close. As the CVs close, reactor pressure (and hence throttle pressure) will begin to rise. This will be seen by the “B” regulator. So as the “A” side pressure error signal goes down, the “B” pressure error will go up. This will eventually cause the “B” regulator to take over at a slightly elevated reactor and throttle pressure

A. Incorrect: As the “A” Side pressure error signal goes down, the “B” pressure error signal will go up. This will eventually cause the “B” regulator to take over and control pressure slightly higher. No scram will occur

B. Incorrect: Due to the -5 psig bias the “B” regulator will take over at a slightly elevated reactor and throttle pressure

C. Incorrect: As the “A” Side pressure error signal goes down, the “B” pressure error signal will go up. This will eventually cause the “B” regulator to take over and control pressure slightly higher. No scram will occur

This would occur if the “A” regulator failed high

D. Correct: As the “A” steam throttle pressure falls, the “A” side pressure error signal will go down. This will cause the CVs to begin to close. As the CVs close, reactor pressure (and hence throttle pressure) will begin to rise. This will be seen by the “B” regulator. So as the “A” side pressure error signal goes down, the “B” pressure error will go up. This will eventually cause the “B” regulator to take over at a slightly elevated reactor and throttle pressure

Technical Reference(s):	AOP-262 rev 10 Automatic Actions page 2 SD-693.2A EHC Logic rev 7 (61) Figure 17 EHC Logic Control System page	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	52.01.01.02 Given an EHC system operating mode and various plant conditions, predict how the EHC system will be impacted by failures in the following support systems: a. 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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	5
	55.45	3

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000 A3.05	
	Importance Rating	3.0	

245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary: A3.05 - Ability to monitor automatic operations of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS including: Control valve operation. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 60

The plant is operating at 100% electrical output when the main turbine speed rises to 1910 RPM. Keeping your response to the Turbine Control system only, which of the following is CORRECT for the listed turbine valves?

	Control Valves	Bypass Valves	Intercept Valves
A.	OPEN	CLOSED	CLOSING
B.	CLOSED	OPEN	OPEN
C.	CLOSED	OPEN	CLOSING
D.	CLOSED	CLOSED	OPEN

Proposed Answer: C

Explanation: Based on either the EHC Logic Control System diagram on page 61 of SD-693.2 Rev… or the Control Valve Intercept Valve Sequencing diagram on page 33 of SD-693.2… show the sequence of the turbine valves. From 1800 RPM to 1890 RPM the Turbine Control Valves ramp close, then at greater than 1890 RPM the intercept valves begin to ramp close to protect the turbine from overspeed. From the EHC logic control system diagram, in this condition the rest of the response is that the Bypass Valves will continue to maintain RPV pressure.

A. Incorrect: Control valves close from 100% to 105% speed (1800-1890 RPM) Bypass valves will be maintaining pressure

B. Incorrect IVs will ramp closed from 105% to 107% (1890 to 1926 RPM)

C. Correct

D. Incorrect, Bypass valves will be maintaining pressure,
IVs will ramp closed from 105% to 107% (1890 to 1926 RPM)

Technical Reference(s): SD 693.2A rev 7 Overspeed and overpressure conditions (page 32-33) explanation and graph (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	51.00.00.04 Describe the operation of the following principle main turbine system components: d. Main stop valves e. Control valves f. Combined Intermediate valves	(As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000 A2.14	
	Importance Rating	3.3	

256000 (SF2 CDS) Condensate: A2.14 - Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low Condensate storage tank level. (CFR: 41.5 / 45.6)

Proposed Question: RO Question 61

The plant is operating at 95% reactor power when the CST level lowered to the low level alarm set point.

(1) What is the operational impact at this level in the CST

AND

(2) How is the CST level restored?

A. (1) 1P-12A and B Condensate Service Water Pumps and 1P-11 Condensate Service Water Jockey Pump TRIP
(2) Transfer Demin Water to the CSTs

B. (1) 1P 12-A and B Condensate Service Water Pumps and 1P-209 A and B CRD Pumps TRIP
(2) Align Core Spray to the CST

C. (1) 1P-12A and B Condensate Service Water Pumps and 1P-11 Condensate Service Water Jockey Pump TRIP
(2) Align Core Spray to the CST

D. (1) 1P 12-A and B Condensate Service Water Pumps and 1P-209 A and B CRD Pumps TRIP
(2) Transfer Demin Water to the CST

Proposed Answer: A

Explanation:Low CST level at 6Ft will cause an automatic trip of 1P-12A and B the Condensate Service Water Pumps, and 1P-11 Condensate Service Water Jockey Pump
1P-209A/B do not trip on low CST level
Align to the CSTs is incorrect, this is used to fill the Torus. Torus height is not sufficient to fill the CSTs

Demin water storage tank 1T45 is the normal makeup source to the CSTs via a Makeup Demin trailer and 1P13A/B demin water transfer pumps.

A. Correct

B. Incorrect, Aligning core spray is incorrect. This is used to fill the Torus not the CST. The student may incorrectly assume Torus height is sufficient to effect the same method of filling the Torus from the CST

C. Incorrect

D. Incorrect. The CRD pumps do not TRIP from a low level in the CSTs, the suction head to the CRD pumps is provided by the reject line from Condensate Discharge. This was not provided to the student as a condition in the stem.

Technical Reference(s):	ARP 1C06A (B-8 and B-9) rev 81 Automatic actions page 48 & 50 OI-537 Condensate/Demin Service Water Section 6.0 Filling the CSTs rev 54 (page 16)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	45.01.01.01 relate the precautions and limitations, operating cautions, or procedural notes of OI-639, and any applicable ARPs to any component or feed and condensate system operating status.	(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
	55.45	6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 A1.06	
	Importance Rating	2.7	

259001 (SF2 FWS) Feedwater: A1.06 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: Feedwater heater level. (CFR: 41.5 / 45.5)

Proposed Question: RO Question 62

Feedwater heater 1E-6A, CV1151, Dump valve has failed open, which of the following is the predicted change in the following?

(1) Final Feedwater temperature will ___(1)___.

AND

(2) Reactor Power will ___(2)___.

A. (1) lower
(2) lower

B. (1) lower
(2) raise

C. (1) raise
(2) lower

D. (1) raise
(2) raise

Proposed Answer: B

Explanation:CV-1151 the 1E-6A feedwater heater dump opening would divert reheat steam from the feedwater heaters and dump it to the condenser, lowering final feedwater temperature Lower feedwater temp will cause its density to increase (more H20) molecules allowing more neutron moderation. This will allow more thermal neutrons and further fission and increased power .

A. Incorrect: temp will lower but power will rise. If the student misinterprets the direction of power change they may pick this choice.

B. Correct

C. Incorrect: high pressure feedwater heater dump opening and removing extraction steam from the feedwater process will lower FW heating via dumping it to the condenser. Power will rise not lower due to more neutron attenuation and fission

D. Incorrect: Final feedwater temperature will not rise it would lower per discussion above. Reactor power will

Technical Reference(s):	AOP-646 Loss of Feedwater heating rev 25 Immediate Operator Action, lower reactor power less than 1912 MWTh (page 2)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	46.00.00.07 evaluate plant conditions and control room indications to determine if the extraction steam and feedwater heating system is operating as expected, and identify any actions that may be necessary to place the extraction steam and feedwater hearing system or the plant in the correct condition.	(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
	55.45	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 K6.11	
	Importance Rating	3.2	

271000 (SF9 OG) Offgas: K6.11 - Knowledge of the effect that a loss or malfunction of the following will have on the OFFGAS SYSTEM: Condenser vacuum.
(CFR: 41.7 / 45.7)

Proposed Question: RO Question 63

How will excessive Main Condenser air inleakage affect operation of the Offgas System?

Offgas process flowrate will rise and ___________.

A. result in recombiner temperatures exceeding 875°F high temperature limit

B. CV-4108, Offgas Outlet Isolation, will isolate

C. CV-4151, Offgas Steam Jet Air Outlet, will isolate

D. Offgas Loop Seals will isolate

Proposed Answer: D

Explanation: Offgas isolation signals are
>/= 4 psig at inlet to SJAE hold up line➔ isolates all loop seal drains (CV-1379, 4126, 4179, 4106, 4107A/B)
.>/= 4.5 psig at inlet to 30 minute holdup line➔ isolates all loop seal drains
</= 4000 lbm/hr motive steam flow to offgas jet compressor AND ,/= 270 psig steam supply pressure to the offgas jet compressor➔ closes MO-4151 offgas jet compressor inlet valve (MO position interlock closes CV-1379 and CV-4126 and isolate HWC.

A. Incorrect: Offgas recombiner temperatures would lower not rise in this case. Student may apply assume recombination would increase in this case raising recombiner temperatures.

B. Incorrect: CV-4108 Offgas Outlet Isolation no longer has automatic isolation signals.To ensure scram frequency reduction, CV-4108 fails open on a loss of instrument air or electrical power to the valve. A valve position interlock from CV-4108 causes all loop seals to isolate when CV-4108 closes.

C. Incorrect: CV-4151 does not have an isolation

D. Correct: Excessive air inleakage would cause system pressure to rise and isolate the loop seals

Technical Reference(s): ARP 1C34 (C-5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	47.01.01.14 evaluate plant conditions and control room indications to determine if the Offgas and recombiner system is operating as expected, and identify any actions that may be necessary to place the Offgas and recombiner 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

55.45 7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000 K4.06	
	Importance Rating	3.4	

286000 (SF8 FPS) Fire Protection: K4.06 - Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Fire suppression capability that does not rely on the displacement of oxygen (Halon): Plant-Specific. (CFR: 41.7 / 45.7)

Proposed Question: RO Question 64

The Jockey Fire Pump, 1P-47, fails and fire main header pressure is lowering slowly. Which one of the following is the first expected automatic response of the fire protection system?

The ____(1)____ Fire Pump will initiate at ___(2)___ psig.

A. (1) 1P-49, Diesel
(2) 85

B. (1) 1P-49, Diesel
(2) 95

C. (1) 1P-48, Electric
(2) 85

D. (1) 1P-48, Electric
(2) 95

Proposed Answer: D

Explanation: Jockey fire pump normally maintains fire header pressure and cycles between psig120 psig and 130 psig
1P48 the electric fire pump starts at a system pressure of 95 psig
1P49 the diesel fire pump starts at a system pressure of 85 psig

A. Incorrect : although this is the correct pressure for the diesel fire pump auto start the electric would start at 95 psig and prevent system pressure from lowering to 85
B. Incorrect: The electric fire pump 1P48 starts at 95 psig
C. Incorrect: the electric fire pump starts at 95 psig. Listed pressure is for the 1P49 diesel fire pump auto start
D. Correct: 1P48 electric fire pump would start at 95 psig

Technical Reference(s): ARP 1C40 (rev 77) J-1 and J-5 (page 120 and 130 of 133 and) (Attach if not previously provided)

OI-513 Fire Protection rev 139
(page 21and 22 of 123 Note)

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 7

55.45 7

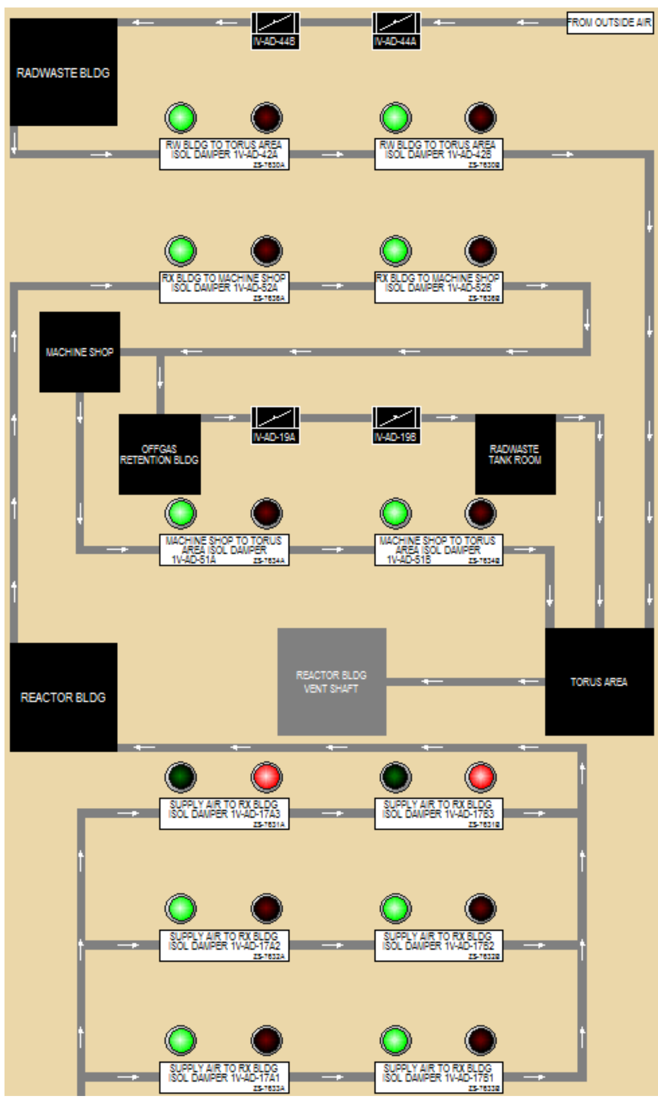
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	288000 K3.04	
	Importance Rating	3.2	

288000 (SF9 PVS) Plant Ventilation: K3.03 - Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on following: Secondary containment pressure. (CFR: 41.5 / 45.3)

Proposed Question: RO Question 65

The plant is operating at 100% reactor power and receives a valid Group 3 signal.

During verification of the Group 3 isolation the following is observed at 1C23:



What, if any, could be the impact of this indication?

A. Unmonitored ground level release

B. SBGT train inlet relief valve will open

C. Reactor Building air temperature will lower

D. No impact this is normal for a Group 3 Isolation

Proposed Answer: A

Explanation: LI-AA-102-1001 Regulatory Reporting. Attachment 1 page 5 of 8. 8 Hour Report 50.72.(b)(3)(v) Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems needed to:

- Shut down the reactor and maintain it in a safe shutdown condition
- Remove residual heat
- Control the release of radioactive material, or
- Mitigate the consequences of an accident
-

A. Correct
Failure of both isolation dampers in one line could lead to unmonitored release

B. Incorrect: SBGT inlet relief would lift due to venting Primary Containment with a high pressure present

C. Incorrect: Group 3 isolation will cause reactor building supply and exhaust fans to trip and will cause building temperatures to rise.

D. Two dampers within the same penetration have failed to close, this is abnormal and not an expected result

Technical Reference(s):	IPOI-7 Group 3 verification LI-AA-102-1001 Regulatory Reporting	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	50007.05.02 Explain the isolation signals with respect to setpoints, components affected and the reason for each 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		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	5	
	55.45	3	

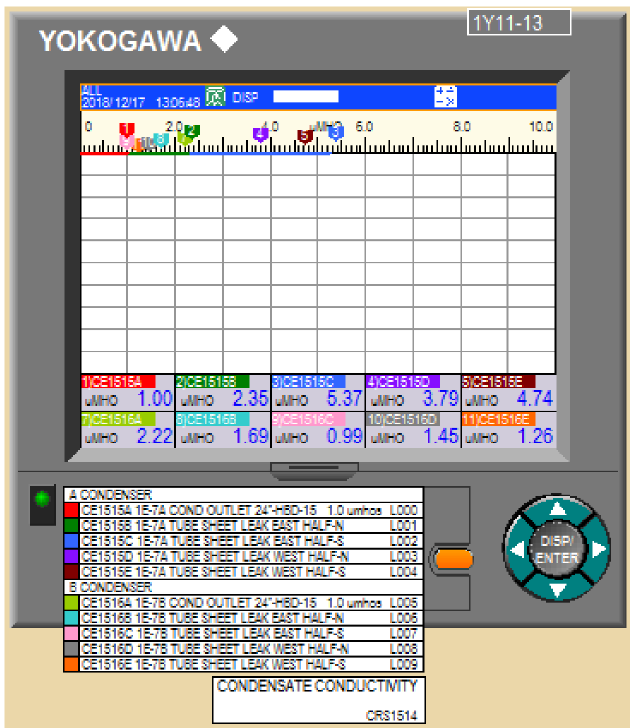
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.34	
	Importance Rating	2.7	

Knowledge of primary and secondary plant chemistry limits. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: RO Question 66

The plant is operating at 50% reactor power when a conductivity transient is observed at 1C06.

The following is observed:



Which of the following action, if any, would an Operator take?

A. Insert a manual reactor scram

B. Perform a fast power reduction

C. Close Main Steam Isolation Valves

D. No action is required at these values

Proposed Answer: A

Explanation: AOP-639 Reactor Water/Condensate High Conductivity
Follow up action 9
If a valid condensate demineralizer influent conductivity reading of greater than or equal (>/=) 1.0 umho/cm (CRS-1514 Pt 1 or 7 at 1C06 or CIT 1704 at 1C80) is received and is not due to condensate system startup or plant startup
Then Reduce reactor recirc flow to 39 Mlbm/hr per IPOI-4, Fast Power Reduction AND Manually Scram the reactor and carry out IPOI-5 Reactor Scram

A. Correct: At >/= 1.0umho/cm (current value is 1.5) a reactor scram is required.

B. Incorrect: At 50% power we'd be below 39Mlb/hr core flow. Student may incorrectly assume a fast power reduction should occur as it would from full power

C. Incorrect: This action is required later in the AOP at a value of >/= 5.0 umho/cm conductivity

D. Incorrect: at this value a reactor scram is required. Student may confuse the 1 and 5 umho/cm limitations

Technical Reference(s):	AOP-639 Reactor Water/Condensate High Conductivity rev 38 (page 4)	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.18.04.01 evaluate plant conditions and control room indications and determine if a fast power reduction or reactor scram is warranted	(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
	55.43/45	5/12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.3	
	Importance Rating	3.7	

Knowledge of shift or short-term relief turnover practices. (CFR: 41.10 / 45.13)

Proposed Question: RO Question 67

With the plant at full power, the following conditions exist:

- You are an NSOE coming in to start your first day of work
- You last stood watch 96 hours ago
- You expect to be working in the Work Control Center, unassigned to the shift

Four hours into the shift:

One of the Control Room NSOEs is required to attend a briefing, and you have been asked to relieve him for about 1 hour.

Which of the following identifies:

(1) how far back you are required to read the Station Log prior to taking the watch

AND

(2) whether or not the NG-016K, NSOE and ANSOE Turnover Form, must be used?

A. (1) 72 hours
(2) The Turnover form must be used.

B. (1) 72 hours
(2) The Turnover form is NOT required.

C. (1) 100 hours
(2) The Turnover form must be used.

D. (1) 100 hours
(2) The Turnover form is NOT required.

Proposed Answer: A

Explanation: Conduct of Operations OP-AA-100-1000 Rev 25. Attachment 8 Shift Relief and Turnover
3.1 The on-coming watchstander shall perform the following

- Review the station log back to the last time the individual stood the watch or three days (whichever is less)
- ACP 1410.10 Shift Turnover / Shift Brief
- Section 3.3 Watchstnader Relief during the shift
- The applicable shift turnover form should be utilized for the watchstanders position being relieved. For shift trunovers requiring initials, N/A should be used when appropriate for the positions not being relieved
-

A. Correct. 1st part correct, 2nd part correct. According to OP-AA-100-1000 Conduct of Operations, the on-coming watchstander shall review the Station Log back to the last time the individual stood the watch or three days (whichever is less). Since the operator last stood watch 96 hours ago, the requirement is to review back for ONLY 72 hours. According to ACP 1410.10 (section 3.3) Watchstnader Relief during the shift the applicable ShiftTurnover Form should be used for the watchstander's position being relieved In the event that only one position (Shift Manager, CRS, STA, NSOE, or ANSOE) is relieved, then an N/A should be placed where appropriate on the shift turnover form for the non-relieved crew members.

B. Incorrect. 1st part correct, 2nd part wrong. This is plausible because the operator may incorrectly believe that the form does not need to be used because the relief is in the middle of a shift and of a temporary nature.

C. Incorrect. 1st part wrong, 2nd part correct. This is plausible because the operator last stood the watch 96 hours ago, and four hours have elapsed since the start of the work day. The operator may incorrectly believe that they are required to read the log back to the time that they last held the shift.

D. Incorrect. 1st part wrong, 2nd part wrong. This is plausible because the operator may incorrectly believe that they are required to read the log back to the time that they last held the shift; and because the operator may incorrectly believe that the form does not need to be used because the relief is in the middle of a shift and of a temporary nature.

Technical Reference(s):	OP-AA-100-1000 Conduct of Operations rev 26 Attachemnt 8 (pages 62) section 3.1 step 12 ACP 1410.10 Shift Turnover/Shift Brief rev 49 Section 3.3 relief during the shift	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	96.05.05.03 Explain the requirements and instructions of ODI-009, "Reactor Operator, Senior Reactor Operator, and Shift Technical Advisor Qualification requirements"	(As available)

Question Source: Bank # X

Modified Bank #		(Note changes or attach parent)
New		

Question History:	Last NRC Exam:	PDA 2009

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

10 CFR Part 55 Content:	55.41	10
	55.45	13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.39	
	Importance Rating	3.6	

Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: RO Question 68

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 210°F in MODE 1

Proposed Answer: C

Explanation: Turbine Trip at 40% reactor power will cause an auto scram
Reactor power spike to 16 % in Mode 2 is above the automatic scram setpoint of 15%
MSIV position automatic scram signal from Group 1 signal at 200°F
EOP-2 T/L-3 wait until torus level cannot be maintained above 7.1 ft (EOP-1 entered) EOP-1 is entered at entry point 1 ensures that, if possible, the reactor is scrammed before ED is initiated.

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 (page 11 of 69) (Attach if not previously provided)

Proposed References to be provided to applicants during examination:		N	
Learning Objective:	96.07.01.20 Explain the requirements for equipment manipulations and control of plant equipment for on-shift and/or off-shift operations personnel	(As available)	
Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	2017 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43/45	5/12	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.14	
	Importance Rating	3.9	

Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10 / 43.3 / 45.13)

Proposed Question: RO Question 69

Which of the following is an approved method to control equipment configuration of the plant?

A. System Descriptions

B. Operating Instruction Lineups

C. Operations Equipment Database

D. Archived clearance order restoration directions

Proposed Answer: B

Explanation: Operating Instruction Lineups are controlled documents and are used to maintain plant configuration control.
OP-AA-101-1000 rev 19 Clearance and tagging, Work complete and tag removal 4.8.3.C clearance removals shall be prepared using controlled references when available
Preparing Clearances The clearance preparer shall prepare a clearance document using available references and/or walkdowns. Controlled references should be used, when available.

A. Incorrect: The system descriptions do contain simplified drawings and are controlled documents however they are not approved for configuration control

B. Correct

C. Incorrect: The Operations equipment database is uncontrolled and can only be used for information only.

D. Incorrect: Archived clearances do contain component identifications and as left configuration however they should not be used to control configuration of plant components

Technical Reference(s): OP-AA-101-1000 Rev 19 page 35 of 193 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	96.07.01.19 Describe the processes by which configuration control is maintained at the station	(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
	55.43/45	3/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.36	
	Importance Rating	3.1	

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)

Proposed Question: RO Question 70

Which of the following associated subsystem would be declared INOPERABLE as a result of placing a Diesel Generator Control handswitch at 1C08 in Pull-to-Lock?

A. Core Spray

B. River Water Supply

C. Residual Heat Removal

D. Emergency Service Water

Proposed Answer: D

Explanation: OI-324 P&L 30
Placing A/B Handswitch HS3231A/B at 1C008 in PTL makes the associated ESW pump Inoperable due to preventing it from Auto starting. Only place in PTL when necessary to prevent equipment damage or personnel injury.

A. Incorrect: The associated Standby Diesel Generator would be INOPERABLE if the Core Spray automatic start logic was inoperable.

B. Incorrect: The associated subsystem would be INOPERABLE if the River Water Supply subsystem was INOPERABLE on the opposite train.

C. Incorrect: This is a supported system by the Standby Diesel Generator, however all features of the Residual Heat Removal subsystem remain OPERABLE.

D. Correct: Placing the A/B SBDG Handswitch (HS3231A/B) at 1C08 in PTL (degraded the power source) makes the associated ESW pump INOPERABLE due to preventing it from auto-starting.

Technical Reference(s): OI 324, Rev. 125 (page 8 of 89) P&L 30 (Attach if not previously provided)

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	10	
	55.43/45	2/13	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.38	
	Importance Rating	3.6	

Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Proposed Question: RO Question 71

Reactor Power is 1900 MWth and rising 1 MWth/minute. At what time will the facility license limit be reached first?

A. 5 minutes

B. 11.5 minutes

C. 12 minutes

D. 15 minutes

Proposed Answer: C

Explanation: Technical Specification Renewed Facility Operating License, DPR-49, Maximum Power Level: Nextera Energy Duane Arnold, LLC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1912 megawatts (thermal). Amendment 306
At current rate of rise 1 MWth/minute the facility can operate 12 minutes prior to reaching its licensed thermal power limit.
IPOI-3 P&L #2 The licensed power limit of 1912 MWTh shall not be intentionally exceeded. Operators shall take prompt action if rated thermal power (RTP) exceeds the limit of 1912 MWTh.

A. Incorrect
Plausible if candidate believes 1905 limit as stated in IPOI-3 P&L 3(f)
Prior to the PPC being shutdown for scheduled maintenance that could last 4 hours or longer, reactor power should be lowered with recirc flow to 1905 mwth. This will allow sufficient margin for maintaining the 8 hr average (C177) for core thermal power below 1912 mwth while the heat balance is unavailable.

B. Incorrect
IPOI-3 1911.5 is not the facility licensed limit.
IPOI-3 P&L 13 states if the 2 hour average exceeds 1911.5 MWth as deter4mined by computer point NSS063, take prompt action to lower reactor power as necessary to prevent the 2 hour average from exceeding 1912 MWt. h

C. Correct
1912 MWth is the facility licensed limit

D. Incorrect
1915 is above the facility licensed limit

Technical Reference(s):	Tech Spec licensed thermal power limit IPOI-3 P&L 2 (page 3) rev 160	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	94.03.01.03 Relate how each step and its performance meets the mitigation strategies of AOP 255.2	(As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	7,10
	55.43/45	1/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.3.7	
	Importance Rating	3.5	

Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)

Proposed Question: RO Question 72

Who can suspend the requirements of the radiation work permit requirements during a station emergency when the Technical Support Center is not manned?

A. Operations Shift Manager

B. Emergency Planning Manager

C. On-shift Health Physics Technician

D. On-shift Nuclear Station Operating Engineer (NSOE)

Proposed Answer: A

Explanation:

A. Correct: During a declared emergency, the Emergency Response Organization (Emergency Coordinator,) is the responsibility of the Operations Shift Manager, when the TSC is not manned. The OSM can suspend RWP requirements prior to responsibility turnover to the TSC.

B. Incorrect: The Emergency Planning Manager, who can be qualified as the Emergency Coordinator, can suspend the RWP requirements. Since, the TSC is not manned, this individual CANNOT suspend RWP requirements.

C. Incorrect: During normal operations, the On-shift Health Physics Technician, will direct which RWP requirements are necessary for given plant conditions.

D. Incorrect: During a declared emergency, the On-shift Nuclear Station Operating Engineer (NSOE) can provide the radiological brief to responding Operators but CANNOT suspend the RWP requirements.

Technical Reference(s): HPP 3101.05, Admin of RWPs, Rev. 61 (Attach if not previously provided)

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	12	
	55.45	10	

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.
(CFR: 41.12 / 43.4 / 45.9)

Proposed Question: RO Question 73

Which of the following is the minimum number of ARMs on the refuel floor which are required to meet the 10CFR70.24 requirements for criticality monitoring?

A. 2

B. 3

C. 4

D. 5

Proposed Answer: A

Explanation: OI-879.2 Area radiation monitoring system.
While the system is in operation, each power supply provides power to 10 channels. At no time will any power supply be turned off except for servicing of that power supply.
To comply with the Code of Federal Regulations 10CFR70.24 part (a)(1), the ARMs on the refuel floor (RM9153, RM9163, RM9164, and RM9178) are to serve as criticality monitors with 2 of the 4 monitors remaining in service at all times. In the event that more than 2 of the monitors must be removed from service at one time, alternate monitoring with an audible alarm must be provided such that at least 2 monitors are always in service. The alarm setpoint of the alternate monitor must be less than or equal to the setpoint of the monitor it is replacing.

A. Correct
Two is the minimum required number of ARM detectors for refuel floor criticality monitoring

B. Incorrect
4 refuel floor ARM detectors are provided, but only 2 are required

C. Incorrect
4 refuel floor ARM detectors are provided, but only 2 are required

D. Incorrect
4 refuel floor ARM detectors are provided, but only 2 are required

Technical Reference(s): OI-879.2 Area Radiation Monitoring System rev 27 P&L 4 (page 3 of 14) (Attach if not previously provided)

AOP-317 rev 109 Note (page 5)

Proposed References to be provided to applicants during examination: N

Learning Objective:	86.04.01.01 Relate the precautions and limitations, operating cautions, or procedural notes of OI-879.2 to any component or ARM system operating status	(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	12
	55.43	4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.12	
	Importance Rating	4.0	

Knowledge of general operating crew responsibilities during emergency operations. (CFR: 41.10 / 45.12)

Proposed Question: RO Question 74

During Emergency Operating Procedure (EOP) implementation, which one of the following describes the proper use of Abnormal Operating Procedures (AOP) during this time?

A. All AOPs are suspended until EOPs are exited

B. The Only AOP actions that can be taken are those directed by EOPs

C. AOP actions are taken as long as they do not contradict EOP actions

D. All AOP actions are required to be completed regardless of the EOP actions

Proposed Answer: C

Explanation: EOPs can be used in conjunction with other operating procedures (OIs, ARPs, AOPs, etc). However, EOPs are higher tier documents and shall direct the primary response to operational transients that require their use. The decision to utilize other approved procedures during EOP execution rests with the Shift Supervisor / Manager. It other procedures are used while executing EOPs, actions specified in these procedures shall not contradict or subvert actions described in EOPs or degrade the operability of equipment critical to EOP strategies.

A. Incorrect
AOPs Can be used in conjunction with EOPs provided they do not conflict with the EOPs as a higher tier document. Since EOPs are higher tier the student may think they are not to be performed concurrently

B. Incorrect
Any AOPs can be used in conjunction with the EOPs as long as they do not conflict with the EOPs. Some AOPs are not directed by the EOPs and should be performed.

C. Correct

D. Incorrect
As stated above only those AOP actions that do not conflict with the EOPs should be performed as plant conditions require.

Technical Reference(s): ACP 1410.1 Operations Working Standards rev 109 (step 7 on page 16 of 28) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	96.06.06.06 For any given plant operating condition when the Emergency Operating Procedures (EOPS) are entered, determine when the lower tier documents may be used.	(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
	55.45	12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.26	
	Importance Rating	3.1	

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: RO Question 75

Which one of the following is required to be present in the Diesel Generator Room until loaded conditions have been established and temperatures stabilized?

A. A charged Fire Hose

B. A portable CO2 Extinguisher

C. A portable Halon Extinguisher

D. A portable Water Mist Extinguisher

Proposed Answer: D

Explanation: When performing a standby diesel generator surveillance test or other manual startup and loading of the SBDG, the plant operator should have a portable water mist extinguisher in the SBDG Room until unit is loaded, operating temperatures have stabilized, and exhaust lagging is not smoking excessively.

A. Incorrect
A fire hose station is located outside the north end between the SBDG's, but not required to be charged and stationed when running a SBDG.

B. Incorrect
CO2 is not required, Dry chemical and water mist extinguishers are listed on PFP-TB-757

C. Incorrect
A halon extinguisher is not required, not listed on the PFP-TB-757

D. Correct

Technical Reference(s): OI-324 Rev123, P&L 24
PFP-TB-757, Rev 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective:	19.01.01.01 Relate the precautions and limitations, operating cautions, or procedural notes of OI-324 to any component or SBDG operating status.	(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	5

EXAM COVER SHEET

Exam Number/Title: PDA OPS 19-1 NRC Exam, Rev. 0			**Trainee Name:**
Training Program: Operations			**Employee ID:**
Course/Lesson Plan Number(s): 60006			
Total Points Possible: 25	**PASS CRITERIA: ≥ 70%**		**Grade: _____/25=_____%**
Graded by:			**Date:**

EXAM REVIEW AND APPROVAL:	
Submitted by Instructor:	**Date:**
Technical Review by:	**Date:**
Approved by Training Supervision:	**Date:**

EXAM RULES

1.	References may not be used during this exam, unless otherwise stated.
2.	Read each question carefully before answering. If you have any questions or need clarification during the exam, contact the exam proctor.
3.	Conversation with other trainees during the exam 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.	Restroom trips are limited and only one examinee at a time may leave.
6.	For exams with time limits, you have **480 minutes** to complete the exam.
7.	The examinee agrees to refrain from discussing the content of the exam until the end of the exam cycle

EXAM 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 Exam Rules stated above. Further, I have not given, received, or observed any aid or information regarding this exam prior to or during its administration that could compromise this exam."

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 exam 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 #	295004	G2.4.41
	Importance Rating		4.6

295004 (APE 4) Partial or Total Loss of DC Power / 6: Generic K/A 2.4.41 - Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)

Proposed Question: SRO Question 1

Given the following:

- The plant has shutdown and is commencing a cooldown following major flooding of the Cedar River
- Reactor Coolant Temperature is 275°F
- 1D10 the Division-1 125 VDC Distribution Panel is de-energized as a result of the flooding
- Output voltage on Division-2 125VDC is 102VDC and stable

`

Which one of the following Emergency Action Levels should be declared?

A. SA4.1

B. SS6.1

C. SS1.1

D. SS3.1

Proposed Answer: D

Explanation:

A. Plausible since this condition could also result in a loss of most of the annunciator power in the plant resulting in this EAL.

B. Plausible since the plant is in a transient and has lost or is close to losing all 125VDC power which supplies annunciator power.

C. Plausible if the candidate assumes the loss of control power supplied by 125VDC could lead to a loss of AC power from protective relaying or breaker control power.

D. Less than 105 VDC bus voltage on BOTH Div 1 and Div 2 125 VDC busses for 15 minutes or longer. With Div1 125VDC Bus deenergized and Div2 125VDC Bus reading less than 105VDC, both DC Electrical sources resulting in the Site Area Emergency.

Technical Reference(s): EPIP FORM EAL-01, Rev. 11
EAL Bases Document System Malfunction Category EBD S, (Attach if not previously provided)

Rev.11 page 24

Proposed References to be provided to applicants during examination:			EAL Board EAL-01 Rev 11N
Learning Objective:			(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43/45	5/11	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA2.03	
	Importance Rating		3.1

295005 (APE 5) Main Turbine Generator Trip / 3: AA2.03 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Turbine valve position. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 2

The plant is operating at 60% power. TBPV testing is in progress. While depressing the test pushbutton for the No. 1 Bypass valve, the valve fails to move.

What is required for this failure?

Declare the No. 1 Bypass INOPERABLE and per _______________.

A. LCO 3.2.2 assess a MCPR limit penalty to ACCUMEN per Reactor Engineering for Bypass Valve INOPERABLE within 4 hours

B. LCO 3.7.7, Restore the Bypass Valve to OPERABLE within 2 hours

C. LCO 3.7.7, No action is required to be taken since only 1 Bypass Valve is REQUIRED to be OPERABLE for this power level and plant configuration

D. LCO 3.7.7, Lower reactor power to < 21.7% power within 2 hours

Proposed Answer: B

Explanation:

A. Incorrect
Required time is 2 hours. This is plausible since some action requirements for similar conditions are to be completed within 4 hours.

B. Correct
LCO 3.7.7 The Turbine Bypass System shall be Operable , Requirements of the LCO not met Satisfy the LCO within 2 hours

C. Incorrect
Tech Spec Bases 3.7.7 An operable main turbine bypass valve requires the bypass valves to open in response to increasing main steam line pressure. No action is plausible since the applicant may believe that 1 TBPV is sufficient for the LCO

D. Incorrect
LCO 3.7.7 required action A.1 Satisfy the requirements of the LCO within 2 hours. Action B.1 if the required action and completion time are not met then B.1 Reduce thermal Power to < 21.7% RTP within 4 hours. This is plausible since the applicant may

determine that we have exceeded the action time of the previous condition.

Technical Reference(s):	Tech Spec 3.7.7 Turbine Bypass System (page 3.7-16) Amendment 243 Tech Spec 3.2.2 Minimum Critical Power Ratio (MCPR) page 3.2-2 Amendment 280	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.2.2 and L3.7.7

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43/45	5/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 G2.2.44	
	Importance Rating		4.4

295018 (APE 18) Partial or Complete Loss of CCW / 8: Generic K/A 2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

Proposed Question: SRO Question 3

While operating at 25% reactor power with the following conditions:

- Stator Water Conductivity has risen to 0.75 μmho/cm and stabilized
- 10 minutes later, the running stator cooling pump “A” TRIPS and will not restart
- Attempts to start Stator Cooling Pump “B” are not successful

What procedure contains the Operator actions for this condition and what action should be directed based upon these conditions?

A. ARP 1C83A B-1 CONDUCTIVITY ABOVE 0.5 MICROMHOS, Continue to monitor conductivity and if conductivity rises to 9.9 μmho/cm, go to ARP 1C83A B-2 CONDUCTIVITY ABOVE 9.9 MICROMHOS

B. ARP 1C83A B-1 CONDUCTIVITY ABOVE 0.5 MICROMHOS, Flush the Stator Water Cooling System until conductivity drops below 0.5 μmho/cm

C. AOP 697 LOSS OF STATOR COOLING. If conductivity rises to 9.9 micromhos/cm, SCRAM the reactor, TRIP the Main Turbine, TRIP the Main Generator.

D. AOP 697 LOSS OF STATOR COOLING, SCRAM the reactor, TRIP the Main Turbine, TRIP the Main Generator

Proposed Answer: D

Explanation: AOP 697 requirement to scram if conductivity is initially >0.5 μmho/cm when loss of all Stator Water Cooling Pumps occurs
Conductivity prior to flow loss was > 0.5 μmho/cm. Per AOP caution immediate scram is required
OI-697 P&L 10.c If stator cooling conductivity is greater than) 0.5 michrmho/ cm prior to flow stoppage, the turbine shall be tripped, generator removed from the grid and field breaker opened as soon as possible. The conductivity of the stagnant water will increase rapidly dur to operation of the generator.

A. Incorrect: Plausible if candidate fails to realize the AOP 697 requirement to scram if initially >0.5 μmho/cm when loss of all Stator Water Cooling Pumps occurs

B. Incorrect: This is plausible since flushing the system could lower conductivity however given the conditions in the stem this is not a viable option and does not address the causal factors. No requirement to flush to < 0.5 μmho/cm in the ARP

C. Incorrect: The Conductivity prior to flow loss was > 0.5 μmho/cm. Per AOP caution immediate scram is required. Plausible if the applicant does not identify the requirement to scram due to the conditions in the stem.

D. Correct: AOP 697 Rev 5. If stator conductivity prior to loss of flow is greater (>) 0.5 umho/cm. the turbine shall be tripped, the generator removed from the grid, and the field breaker opened within 3 minutes

Technical Reference(s):	AOP-697 Rev 5 Loss of Stator Cooling (CAUTION on page 2) OI-697 Rev 55 P&L 10.c (page 4 of 20)	(Attach if not previously provided)

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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	5
	55.43/45	5/12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EA2.02	
	Importance Rating		4.0

295024 High Drywell Pressure / 5: EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell temperature. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 4

The crew has just completed a Fast Power Reduction to 78% power after a steam leak in the drywell was detected.

Current plant conditions:

- Drywell air temperature is 180°F and rising 1°F/ 1 minute
- Drywell pressure is being maintained between 1.0 and 1.5 psig through venting
- The crew is briefing continuation of plant shutdown by inserting control rods

What procedure directs the actions the crew takes to address these conditions and what action should be taken next?

A. IPOI 4, Reactor Shutdown - Insert Control rods to get below 70% power

B. AOP 573, Primary Containment Control - Insert a Manual Reactor SCRAM

C. EOP 2, Primary Containment Control - Anticipate Emergency Depressurization per SEP 307

D. EOP 3, Secondary Containment Control - Continue controlled plant shutdown to cold shutdown conditions

Proposed Answer: B

Explanation: AOP-573 Follow up action 6 and 7 if drywell pressure cannot be maintained <1.5 psig or Drywell temperature cannot be maintained <180°F THEN reduce reactor power per IPOI-4 Fast power reduction to restore and maintain within limits.
If DW pressure still cannot be maintained <1.5 psig or DW temperature cannot be maintained <180°F THEN Manually SCRAM

A. Incorrect: AOP 573 states that following a fast power reduction, if Drywell air pressure rises to 1.5 psig or DW air temperature rises above 180°F a reactor scram is required. Plausible if the applicant does not determine that power has already been reduced via fast power reduction and the direction at this point is to SCRAM.

B. Correct: AOP 573 Step 7 states If DW pressure still cannot be maintained < 1.5 psig OR DW Temp cannot be maintained <150°F Then manually Scram.

C. Incorrect: EOP 2 entry is met being above 150°F. However anticipate ED should not be performed until the reactor is shutdown. Plausible if the applicant fails to implement the step to SCRAM prior to reducing pressure

D. Incorrect: The conditions described in the stem contain No EOP 3 entry conditions. EOP 3 is not applicable based upon these indications. Plausible if the applicant incorrectly applies the guidance for secondary containment control vice primary containment control.

Technical Reference(s): AOP-573 Rev 7(page 4 of 8)
IPOI-4 Rev 141 (Attach if not previously provided)

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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43/45 5/13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025	G2.2.40
	Importance Rating		4.7

295025 (EPE 2) High Reactor Pressure / 3: Generic K/A 2.2.40 - Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)

Proposed Question: SRO Question 5

- 10:00 While performing a plant walkdown with the plant at 85% reactor power, the Balance of Plant (BOP) Operator reports that the “B” EHC Pressure Regulator is in service
- 10:02 The At the Controls Operator reports that Reactor Pressure is 1030 psig
- 10:17 Attempts to lower Reactor Pressure were unsuccessful

(1) The CRS will direct _________________ to mitigate this condition?

AND

(2) What is the first action that must be completed?

A. (1) AOP 693 Main Turbine/EHC Failures
(2) Be in MODE 3 within 12 hours

B. (1) AOP 693, Main Turbine/EHC Failures
(2) Reduce Thermal Power to <21.7% within 4 hours

C. (1) AOP 262, Loss of Reactor Pressure Control
(2) Be in MODE 3 within 12 hours

D. (1) AOP 262, Loss of Reactor Pressure Control
(2) Reduce Thermal Power to <21.7% within 4 hours

Proposed Answer: D

Explanation: AOP-262 Rev 10 Follow up action 6 If (1) turbine pressure regulator has failed of is out of service and reactor power is <90% of rated THEN reduce reactor power to <21.7% within 4 hours per MCPR and APLHR Technical Specification

A. Incorrect – AOP 693 does not provide the guidance for a failed pressure regulator. The first action required to be completed is thermal power to <21.7% within 4 hours in accordance with LCO 3.2.2. This is plausible if Tech Specs are incorrectly applied and the applicant does not take action to lower power first. This action would eventually be required if efforts to exit the applicability are not successful

B. Incorrect – AOP 693 does not provide the guidance for a failed pressure regulator. Plausible if the student does not know the governing procedure for the given conditions. While AOP 693 provides EHC system guidance it does not provide the guidance for inoperable pressure regulators.

C. Incorrect – The first action required to be completed is thermal power to <21.7% within 4 hours in accordance with LCO 3.2.2. Plausible if the student does not implement the first action required to be taken.

D. Correct – AOP 262 provides the guidance for a failed pressure regulator. Step 6 directs lowering power <21.7% to comply with the MCPR spec 3.2.2.

Technical Reference(s):	AOP 262 Loss of Reactor Pressure Control, rev. 10 step 6,(page 4) LCO 3.2.2 MCPR rev. 244, LCO 3.4.10 Reactor Steam Dome Pressure Rev. 224	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: L3.2.2 and L3.4.10

Learning Objective: (As available)

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

Question History: Last NRC Exam: PDA 17-1

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

10 CFR Part 55 Content:	55.41	10
	55.43	2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EA2.01	
	Importance Rating		4.2

295026 (EPE 3) Suppression Pool High Water Temperature / 5: EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 6

The following conditions are present following a Reactor SCRAM:

- Reactor pressure is 910 psig and stable
- Torus water level is 10.4 feet and rising slowly
- Torus water temperature is 160°F and rising slowly
- Drywell pressure is 6 psig and rising slowly
- Drywell air temperature is 220°F and rising slowly

The Control Room Supervisor will direct ____________.

A. vent the Drywell

B. emergency depressurization

C. lowering reactor pressure to 200 psig

D. anticipate emergency depressurization

Proposed Answer: B

Explanation: Heat Capacity Limit Graph 4 has been exceeded. Emergency Depressurization is required. Prior to exceeding Graph 4 EOP-1 PC/P-1 Continuous Recheck statement allows lowering RPV/pressure. If torus water temperature cannot be maintained below the Heat Capacity Limit (Graph 4) and reducing RPV pressure will not result in loss of injection required for adequate core cooling Then maintain RPV pressure below the limit (OK to exceed cooldown rate limit).
EOP-2 T/T-6 Wait until torus water temperature and RPV pressure cannot be maintained below the Heat Capacity limit (Graph 4)➔ Emergency RPV Depressurization is Required.

A. Incorrect: A containment isolation signal precludes the Operators from venting primary containment. Plausible since venting the containment is a viable mitigating action if the student does not recognize the isolation precludes this action.

B. Correct: As a result of exceeding the Heat Capacity Temperature Limit, EOP 2 requires an Emergency Depressurization when Torus water temperature cannot be maintained

below the Heat Capacity Limit.

C. Incorrect
Graph 4 is exceeded. Emergency Depressurization is required. This would be true if only HPCI or RCIC steam drive systems were available for RPV injection and adequate core cooling. No conditions were provided that AC driven injections pumps were not available.

D. Incorrect,
Prior to exceeding the Heat Capacity Graph 4, allowances are provided to maintaining reactor pressure below the graph. Once the graph is exceeded Emergency Depressurization is required.

Technical Reference(s): EOP 2 Bases, Rev.16 page 29 (Attach if not previously provided)
EOP 1 RPV Control step RC/P-2, Rev.20
EOP-2 T/T-6 Rev 18

Proposed References to be provided to applicants during examination: EOP 2 Graph 4

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000	G2.1.20
	Importance Rating		4.6

700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6: Generic K/A 2.1.20 - Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Proposed Question: SRO Question 7

The plant is operating at 100% reactor power. The balance of plant operator notices the following from the 1C08 panel:

- Generator MVARS are 200 **LAGGING**
- Essential Bus Voltages 4120V and lowering slowly

The ITC has contacted the control room and informed them that storms in the region are expected to impact grid supply and that the grid is entering a limited reserve condition. 161KV Grid Voltage is expected to lower to 159.5KV or 99.07% DAEC post trip contingency voltage.

Based upon these reports and conditions, what should the CRS direct?

A. Enter AOP 304 GRID INSTABILITY, Declare ONLY the Startup Transformer inoperable in accordance with Technical Specification LCO 3.8.1 Condition A

B. Enter AOP 304 GRID INSTABILITY, Declare both offsite sources inoperable in accordance with Technical Specification LCO 3.8.1 Condition C.

C. Enter AOP 301 LOSS OF ESSENTIAL ELECTRICAL POWER, Declare both offsite sources inoperable in accordance with Technical Specification LCO 3.8.1 Condition C

D. Enter AOP 301 LOSS OF ESSENTIAL ELECTRICAL POWER, Declare ONLY the Startup Transformer inoperable in accordance with Technical Specification LCO 3.8.1 Condition A

Proposed Answer: B

Explanation: Any potential loss of a grid component (a “contingency”) other than the DAEC, which would lead to an undervoltage condition in the DAEC switchyard does NOT result in Technical Specification LCO actions.
DAEC actual 161KV switchyard voltages </= 99.2% when 1A3 and 1A4 are connected to the Startup Transformer 1X3 or 345 KV Switchyard voltages </= 98.2% when 1A3 and 1A4 are connected to the Standby Transformer 1X4 may not be an indication of inoperability of offsite power. A trip of the DAEC could potentially cause actual switchyard voltage to increase above minimum value in which case offsite power may be inoperable.
If notified by ITC Midwest that contingency trip of the DAEC would lead to an undervoltage

<u>condition of </= 99.2% in the DAEC switchyard 161KV bus when 1A3 And 1A4 are connected to Startup Transformer 1X3. OR ITC Midwest confirms a trip of the DAEC would lead to an undervoltage condition of </= 98.2% in the DAEC switchyard 345 KV bus when 1A3 and 1A4 are connected to Standby Transformer 1X4. THEN Declare both Offsite Sources inoperable and enter TS LCO action as required by the mode of applicability.</u>

A. Incorrect: No loss of essential power has ooccurred. 1A3/4 would trip on degraded voltage of 91.3% for 8.5 seconds (3798 volts) or SU or SB transformer output voltages of 65% (2704 volts.) Student may select this choice if they do not understand the tech spec requirement.

B. Correct: If notified by ITC that the contingency trip of DAEC would lead to an undrevoltage condition of < 99.2% in the DAEC switchyard 161KV bus when 1A3 and 1A4 are connected to the SU Xfmr then declare both offsite sources inoperable and enter TS LCO as required.

C. Incorrect: No loss of essential power has ooccurred. 1A3/4 would trip on degraded voltage of 91.3% for 8.5 seconds (3798 volts) or SU or SB transformer output voltages of 65% (2704 volts.) Student may select this choice if they do not understand the tech spec requirement to declare both sources inop.

D. Incorrect: If notified by ITC that the contingency trip of DAEC would lead to an undrevoltage condition of < 99.2% in the DAEC switchyard 161KV bus when 1A3 and 1A4 are connected to the SU Xfmr then declare both offsite sources inoperable and enter TS LCO as required. Student may select this choice if they do not understand the tech spec requirement to declare both sources inop.

Technical Reference(s): AOP-304 Rev 51 Grid Instability
Follow up Action 1 page 3
LCO 3.8.1.C Amendment 270 (Attach if not previously provided)

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

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002 AA2.02	
	Importance Rating		3.3

295002 (APE 2) Loss of Main Condenser Vacuum / 3: AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Reactor power: Plant-Specific. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 8

The plant was operating at 78% power following a fast power reduction per IPOI 4 due to rising condenser back pressure. The RO notices the following conditions

- Condenser Back Pressure has risen to 8.5”Hg and continues to rise 0.2”Hg/minute
- Main Turbine has tripped
- MSIVs are closed
- Reactor pressure and power are fluctuating widely with SRV and SV actuation
- Drywell pressure is 5 psig and rising
- Torus water temperature is 85°F and rising

What procedure(s) and actions should the CRS direct FIRST to address these conditions?

A. AOP 693 MAIN TURBINE / EHC FAILURES

B. EOP 1 RPV CONTROL and transition to ATWS

C. EOP 2 PRIMARY CONTAINMENT CONTROL

D. AOP 691 CONDENSER HIGH BACKPRESSURE

Proposed Answer: B

Explanation: ATWS indications are present. Power and pressure fluctuating with SRV and SV actuation. Power is not expected to fluctuate with SRV / SV actuation.
EOP-1 is entered (Scram required with power above 5% or unknown) CRS IF any rod is withdrawn past position 00 AND it has not been determined that the reactor will remain shutdown under all conditions without boron THEN Enter ATWS

A. Incorrect: This is expected to eventually occur however this action is directed by EOP ATWS under the Q leg of ATWS at step Q-7.

B. Correct: EOP 1 and ATWS provide the proper guidance necessary to deal with the conditions provided.

C. Incorrect: EOP 2 alone does not provide the guidance to address the given conditions of an ATWS.

D. Incorrect: While this procedure does provide guidance to deal with the rising condenser backpressure condition, there are other matters for the crew to address that are more pressing, controlling reactor power under the ATWS EOP

Technical Reference(s): EOP-1, Rev 20
ATWS EOP Rev 23 (Attach if not previously provided)

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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022	G2.1.27
	Importance Rating		4.0

295022 (APE 22) Loss of Control Rod Drive Pumps / 1: Generic K/A 2.1.27 - Knowledge of system purpose and/or function.

Proposed Question: SRO Question 9

The plant is operating at 80% reactor power. The crew is withdrawing control rods to raise reactor power when the running "A" CRD Pump trips.

What procedure and action should be taken to address this condition?

A. Place the MODE Switch in SHUTDOWN IMMEDIATELY

B. Utilize applicable steps of AIP 407 MAXIMIZE CRD INJECTION to restore CRD flow

C. The Annunciator Response Procedure will direct the operator to OI 255 to place the "B" CRD Pump in service

D. Enter and follow the actions of AOP 255.1, Control Rod Movement/Indication Abnormal to place the "B" CRD Pump in service

Proposed Answer: C

Explanation: : ARP 1C05A Rev 90 (A-6 / A-7) A/B CRD Pump Trip directs restoration per OI-255.

A. Incorrect: A misinterpretation of the Technical Specifications could lead to this choice 3.1.5, Control Rod SCRAM Accumulators, Condition B, however the conditions in the STEM do not support this action.

B. Incorrect: This is to maximize CRD flow and steps raise cooling and drive water flow and Dp. May drift control rods into the core, requiring the operator to insert a manual scram.

C. Correct: ARP 1C05A Rev 90 (A-6 / A-7) A/B CRD Pump Trip directs restoration per OI-255.

D. Incorrect: This AOP does not provide instruction on placing a CRD pump in service. If student is not familiar with the direction, they may choose this distractor.

Technical Reference(s): ARP 1C05A Rev 90 (A-6 / A-7) A/B CRD Pump Trip (Attach if not previously provided)

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

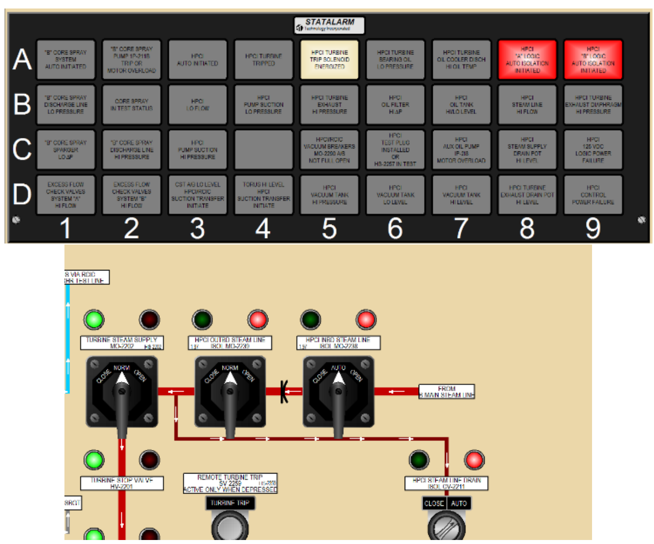
Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295032	EA2.02	
	Importance Rating			3.5

295032 (EPE 9) High Secondary Containment Area Temperature / 5: EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Equipment operability. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 10

The plant is operating at 100% reactor power when the following occurs:

- Steam Leak Detection high ambient temperature
- The Operator notices the following at 1C03 and attempted to manually CLOSE MO2238 HPCI INBD STEAM LINE ISOLATION and MO2239 HPCI OUTBD STEAM LINE ISOLATION
- HPCI Room temperature is 185°F and rising



Based upon the following conditions answer the following:

1) What is the next required action?

AND

2) What is the basis for this action?

A. 1) SCRAM the reactor and enter IPOI 5
   2) To ensure equipment necessary for the safe shutdown of the facility will not fail

B. 1) Emergency Depressurize when the same parameter exceeds its max safe operating limit in 2 or more areas
   2) To ensure equipment necessary for the safe shutdown of the facility will not fail

C. 1) SCRAM the reactor and enter IPOI 5
   2) To ensure the HPCI System remains operable

D. 1) Emergency Depressurize when the same parameter exceeds its max safe operating limit in 2 or more areas
   2) To ensure the HPCI System remains operable

Proposed Answer: A

Explanation: EOP 3 Bases SC-5 Before any parameter reaches its MAX SAFE Operating Limit➔ Scram
MSOL are defined as the highest parameter value at which neither (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. Room temperature is at 185°F and rising already above safe personnel entry.

A. Correct: Step SC-2 has failed Isolate all systems discharging into the area
Max Normal Temp has been exceeded and continues to rise.
Step SC-4 Will RPV pressure reduction decrease leakage into secondary containment (Yes) HPCI is a primary system. So action at SC-5 is Scram prior to reaching MSOL would be correct. MSOL for HPCI room is 310°F

B. Incorrect, waiting for the same parameter to exceed MSOL in same area is ED Criteria. Student may incorrectly apply this step if previous step not implemented.

C. Incorrect. The HPCI system should be declared inoperable given the conditions in the stem.

D. Incorrect: waiting for the same parameter to exceed MSOL in same area is ED Criteria. Student may incorrectly apply this step if previous step not implemented. The HPCI system should be declared inoperable given the conditions in the stem.

Technical Reference(s): EOP 3, Rev.22 Step SC-5
EOP-3 Bases Rev 13 (page 20) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-3 Table 6

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 G2.4.20	
	Importance Rating		4.3

203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode: Generic K/A 2.4.20 - Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 11

Following Emergency Depressurization, plant conditions are as follow:

- RPV Water Level is -10 inches and stable
- ONLY one RHR Pump is available and is injecting at 4800 GPM
- Suppression pool water level is 7.5 feet and slowly rising
- Suppression pool temperature is 212°F and stable
- Drywell Pressure is 3.3 psig and stable
- Torus Pressure is 3.0 psig and stable

Which one of the following actions is required?

A. Continue RPV Injection Flowrate

B. Place Torus Cooling in service maximized

C. Reduce RHR Flowrate to maintain within the NPSH limits for 1 RHR Pump operation

D. Reduce RHR Flowrate to maintain within the VORTEX limits for 1 RHR Pump operation

Proposed Answer: A

Explanation: Where references to this caution (3) occur, the identified systems should be operated within the NPSH and Vortex limits if possible. If the situation warrants. However, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limit in a particular event should consider Immediate and catastrophic failure is not expected if a pump is operated beyond the NPSH or vortex limit. The undesirable consequences of uncovering the reactor core should thus outweigh the risk of equipment damage.

A. Correct: RPV/L is <+15 inches. Injection should continue to restore adequate core cooling.

B. Incorrect: RPV/L is <+15 inches. Inappropriate to divert RHR injection away from the core. Student may select this choice if they believe cooling the Torus takes precedence to injection.

C. Incorrect: RPV/L is <+15 inches. Inappropriate to divert RHR injection away from the core. While this may prevent damage to the RHR pumps restoring adequate core

cooling takes precedence.

D. Incorrect: RPV/L is <+15 inches. Inappropriate to divert RHR injection away from the core. While this may prevent damage to the RHR pumps restoring adequate core cooling takes precedence.

Technical Reference(s):	EOP Cautions Bases Rev 12 page 18. EOP Bases Breakpoints page 7 of 14 (+15 inch) loss of adequate core cooling through core submergence Top of active fuel	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP CAUTION Figure 19 (NPSH) and Figure 23 (Vortex)

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 A2.05	
	Importance Rating		3.8*

206000 (SF2, SF4 HPCIS) High Pressure Coolant Injection: A2.05 - Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCIS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: D.C. electrical failure: BWR-5,6. (CFR: 41.5 / 45.6)

Proposed Question: SRO Question 12

The plant has experienced an event which required the HPCI system to operate to maintain RPV level. During HPCI operation, a loss of HPCI Inverter power occurs.

What procedure should be directed to address this condition and what is the expected response of the HPCI system based upon this condition?

A. AOP 302.1 Loss of 125 VDC; The HPCI turbine will go to idle speed and remain running

B. AOP 388 Loss of 250 VDC; The HPCI turbine will go to idle speed and remain running

C. AOP 302.1 Loss of 125 VDC; HPCI will trip and cannot be restarted

D. AOP 388 Loss of 250 VDC; HPCI will trip and cannot be restarted

Proposed Answer: A

Explanation: OI-152 Rev 119 Appendix 5 Loss of DC Loads and Their Effect on HPCI Logic. 1D23 ckt 02 HPCI initiating, Loss of power to the HPCI inverter and speed control logic. This will result in HV-2000 closing. The loss of the speed control system places the HPCI turbine in the idle speed condition.
AOP-302.1 Automatic Actions on page 35 Loss of power to HPCI speed control circuit (HPCI will not start, and will stop running)

A. Correct
Loss of 125 VDC 1D23 ckt 02 places HPCI Turbine at idle speed condition.

B. Incorrect
This is a loss of Div 2 125 VDC, not 250 VDC. Student may select this choice if the mistaken control power and logic source.

C. Incorrect
Loss during operation causes idle speed. Loss during standby readiness would prevent HPCI from initiating. Student may select this if they do not understand the plant effect from loss of inverter power.

D. Incorrect
This is a loss of Div 2 125 VDC, not 250 VDC. . Student may select this choice if the mistaken control power and logic source. Student may select this if they do not understand the plant effect from loss of inverter power.

Technical Reference(s): OI-152 Rev 119 Appendix 5 (page 74-77)
AOP-302.1 Rev 59 (Automatic Actions page 35)
(Attach if not previously provided)

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 5
55.45 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000	G2.4.6
	Importance Rating		4.7

211000 (SF1 SLCS) Standby Liquid Control: Generic K/A 2.4.6 - Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 13

HPCI and “A”CRD pump are tagged out for maintenance when a Loss of Coolant Accident **AND** a loss of offsite power has occurred.

The following conditions now exist:

- RPV water level is 105 inches and lowering slowly
- Reactor pressure is 850 psig and lowering slowly
- 1A4 Bus Lockout
- RCIC is operating and injection maximized

For the current plant conditions, what alternate injection procedure should be directed to address the lowering RPV Water Level?

Direct alternate injection systems placed in service per _________.

A. AIP 401, Injection with RHRSW

B. AIP 404, Injection with Fire Water

C. AIP 406, Injection with SBLC

D. AIP 407, Maximize CRD injection

Proposed Answer: C

Explanation: With the current RPV pressure the low pressure systems (Fire Water and RHRSW will be below their shutoff head and will not inject into the RPV).
No CRD pumps are available 1P209A is tagged out of service for maintenance, and 1P209B is lost with the 1A4 Bus Lockout
1P230A SBLC is available powered by 1B34 (1G31, A SBDG)

A. Incorrect
Discharge pressure of RHRSW is 0-270 psig, well below the given RPV pressure. The student may select this since this is an alternate injection source.

B. Incorrect
Fire system pressure is 0-125 psig, well below the given RPV pressure. . The student

may select this since this is an alternate injection source.

C. Correct
1B34 available via 1G31 A SBDG and 1P230A "A" SBLC pump available
Discharge pressure 0-1400 psig

D. Incorrect
1P209A CRD is tagged out for maintenance
1A4 is locked out so 1P-209B will have no power. While the discharge head would be sufficient for injection, the system is unavailable given the stem conditions.

Technical Reference(s): EOP-1 Rev 20
Alternate Injection Systems Table 2A
AOP 301 Rev 75 (page 49 and 50)
1P230A breaker 1B3445
(Attach if not previously provided)

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
Comprehension or Analysis X

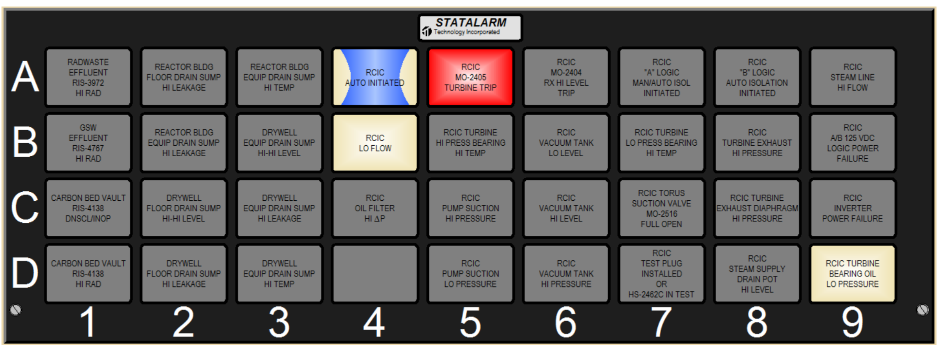
10 CFR Part 55 Content: 55.41 10
55.43 5

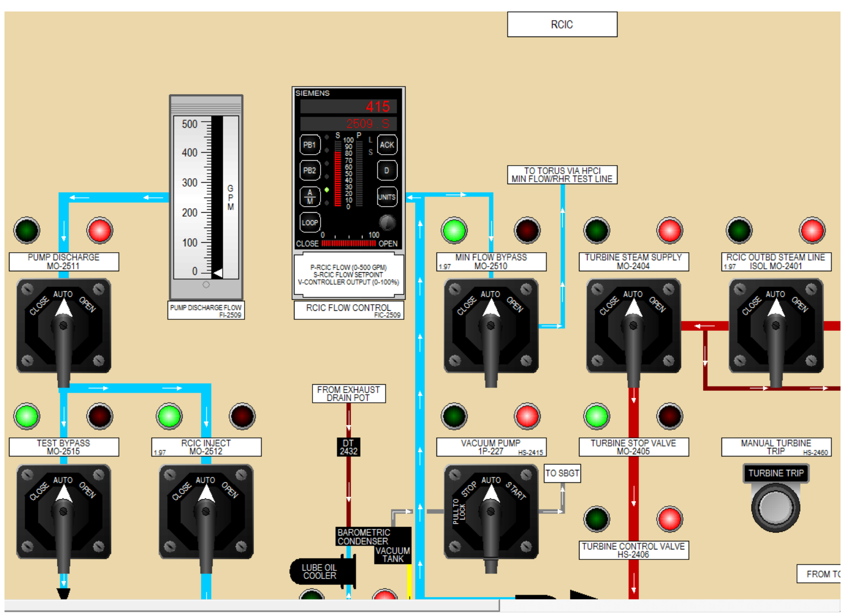
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A2.12	
	Importance Rating		3.0

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling: A2.12 - Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings. (CFR: 41.5 / 45.6)

Proposed Question: SRO Question 14

The plant has experienced a Loss of OFFSITE Power. The BOP Operator is directed to place RCIC in service per the QRC to maintain RPV Water level. While starting RCIC the following is observed:





What Procedure and Action should be directed to start RCIC from this condition?

A. ARP 1C04B A-5 RCIC MO-2405 Turbine Trip. Close MO-2405 and then OPEN MO-2405

B. ARP 1C04B A-5 RCIC MO-2405 Turbine Trip. Close MO-2405. MO-2405 will OPEN Automatically once fully closed

C. SAMP 703 RCIC OPERATION FOLLOWING LOSS OF ELECTRICAL POWER. Close MO-2405 and then OPEN MO-2405

D. SAMP 703 RCIC OPERATION FOLLOWING LOSS OF ELECTRICAL POWER. Close MO-2405. MO-2405 will OPEN Automatically once fully closed

Proposed Answer: A

Explanation: 1C04B (A-5) RCIC MO-2405 Turbine Trip provides guidance to restart RCIC MO2405 does not have an auto re-open

A. Correct: ARP 1C04B A-5 RCIC MO-2405 Turbine Trip provides guidance to restart RCIC

B. Incorrect: ARP 1C04B A-5 RCIC MO-2405 Turbine Trip provides guidance to restart RCIC. MO-2405 will not auto re-open. Student may select if they think MO 2405 will automatically open. It will not.

C. Incorrect: SAMP 703 is to operate RCIC when Div 1 125 VDC batteries are no longer available. Batteries are still available. Student may select this however the SAMP is not authorized for this condition.

D. Incorrect: SAMP 703 is to operate RCIC when Div 1 125 VDC batteries are no longer available. Batteries are still available. Student may select this however the SAMP is not authorized for this condition.

Technical Reference(s): SAMP 703 Rev 8
ARP 1C04B (A-5) rev 85 (Attach if not previously provided)

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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	5
	55.45	6

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262001	G2.2.42	
	Importance Rating			4.6

262001 (SF6 AC) AC Electrical Distribution: Generic K/A 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Proposed Question: SRO Question 15

While operating at 100% reactor power the following annunciator is received:

- 1C08A (C-5), LC 1B3 BREAKER 1B301, 1B302, 1B303, 1B304 TRIP

As a result of the conditions associated with the alarm above, the following equipment loses power:

- 125VDC Battery Charger 1D12
- 250VDC Battery Charger 1D43
- "A" Standby Filter Unit
- ESW Pump 1P99A
- RWCU MO2700

Which one of the following is correct regarding TS actions required to address these conditions?

A. 3.8.1 Condition B, restore the SBDG to OPERABLE status within 7 days

B. 3.7.3 Condition A, restore the ESW subsystem to OPERABLE status within 7 days

C. 3.8.4 Condition A, restore 125VDC Electrical Power Subsystem to OPERABLE status within 8 hours

D. 3.8.7 Condition A restore AC Electrical Power Distribution System to OPERABLE status within 8 hours

Proposed Answer: D

Explanation: TS 3.8.7 bases. The AC and DC electrical power distribution subsystems listed in the LCO are required to be Operable. Based on the number of safety significant electrical loads associated with each bus listed in Table 3.8.7-1, if one or more of the buses become inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Action A.1 the required AC buses, load centers, motor control centers, and distribution panels must be restored to Operable within 8 hours.

A. Incorrect
Important electrical power is lost 1G312, 1V-SF-020 supply fan are lost. Bus 1B32 and or 1B34 must be restored within 8 hours per 3.8.7.A. This may be selected if the student thinks they should cascade. They should not.

B. Incorrect
1B32 is lost as well as 1B34. 3.8.7.A requires restoration within 8 hours. This may be selected if the student thinks they should cascade. They should not.

C. Incorrect
1B32 is lost and 1D12 125VDC charger. 125 VDC distribution is still maintained via battery. TS 3.8.7.still is applicable. This may be selected if the student thinks they should cascade. They should not.

D. Correct

Technical Reference(s): TS 3.8.7 / bases page 3.8-65
(LCO 3.0.6)
LCO 3.0.6 page B 3.0-9
(Attach if not previously provided)

Proposed References to be provided to applicants during examination:
LCO 3.8.7
LCO 3.8.4
LCO 3.7.3
LCO 3.8.1

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7,10
55.43 2-3

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 A2.06	
	Importance Rating		3.1

201003 (SF1 CRDM) Control Rod and Drive Mechanism: A2.06 - Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Suction strainer(s) becoming plugged. (CFR: 41.5 / 45.6)

Proposed Question: SRO Question 16

The plant is operating at 100% reactor power. The previous shift established more frequent monitoring of CRD Suction Filter pressure due to a shift in suction pressure trends. While reviewing Logs, the CRS notices CRD Suction Filter pressures as follows:

00:00:00	9” Hg vacuum
01:00:00	10” Hg vacuum
02:00:00	12” Hg vacuum
03:00:00	14” Hg vacuum

Per OI 255 CRD System, assuming the current trend, what action should the CRS direct for this condition prior to 0500?

A. START the standby CRD Pump

B. Place the standby Suction Filter in service

C. Install jumpers to align CRD Pump suction to the CST

D. Align Condensate Service for CRD Control Rod Assemblies

Proposed Answer: B

Explanation: OI-255 Section 4.0 Normal Operation, CRD suction pressure high limit is listed as 16” Hg vacuum. ARP 1C05A (B-6 / B-7) A/B CRD Pump Lo Suct Pressure, setpoint is 18”Hg. Automatic action is CRD pump trips after 15 second time delay. Crew would place standby filter into service prior to the low suction pressure trip

A. Incorrect: This action would result in an even lower suction pressure. Student may choose if they determine additional CRD flow is necessary to overcome conditions.

B. Correct: see explanation above.

C. Incorrect: This occurs automatically based upon Condensate Pump breaker position. The alignment in the stem precludes this from occurring. Plausible in that this action can occur in the plant.

D. Incorrect: This requires both CRD pumps tagged out and is an outage only option.

Plausible in that this action can be performed in the plant.

Technical Reference(s):	OI-255 rev 97 Section 4.0 Normal Operation ARP 1C05A (B-6/B-7) rev 90 CRD Pump Lo Suction Pressure	(Attach if not previously provided)

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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	5
	55.45	6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 G2.4.30	
	Importance Rating		4.1

202001 (SF1, SF4 RS) Recirculation: Generic K/A 2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

SRO Question 17

The plant is operating at 100% reactor power. A sequence exchange is scheduled to be conducted.

- Reactor Power is required to be lowered to 55%

Who (by title) is required to be informed of this power adjustment?

A. Real time desk and the NDDO

B. Real time desk and NRC resident

C. Operations Director and the NDDO

D. RX Engineering Supervisor and NRC resident

Proposed Answer: A

Explanation: REDP 8 Sequence exchange requirements prerequisites require Real Time Desk notification and NDDO notification. Page 9 of 32 rev 77

A. Correct

B. Incorrect: NRC resident not required for normal power reduction. If a major power adjustment is required NRC resident should be informed.

C. Incorrect: Operations Director not required for normal power reduction. If a major power adjustment is required Operations Director should be informed.

D. Incorrect: RX Engineer developed the plan and the NRC resident not required for normal power reduction

Technical Reference(s): REDP 8 rev 77 (page 9) Prior Reactivity Management Plan with prerequisites page (Attach if not previously provided)

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 10

55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000 K1.04	
	Importance Rating		3.3

234000 (SF8 FH) Fuel Handling Equipment: K1.04 - Knowledge of the physical connections and/or cause-effect relationships between FUEL HANDLING EQUIPMENT and the following: Reactor manual control system. (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Proposed Question: SRO Question 18

The Mode Switch is in REFUEL and all control rods are inserted. The Refueling bridge Operator grappled a fuel bundle, partially raised the grapple, and commenced moving the bundle from the spent fuel pool towards the core.

Which one of the following describes what will result as the Refueling Bridge moves towards the core?

The Refueling Bridge __________.

A. continues over the core and initiates a control rod block

B. continues over the core and causes **NO** other protective actions

C. stops before it reaches the core and initiates a control rod block

D. stops before it reaches the core and causes **NO** other protective actions

Proposed Answer: A

Explanation: Rod Out Block = Fuel Grapple Not Full Up and Refuel Platform Over or Near the Core AND Mode Switch in Refuel
Road Block = Fuel Grapple Not Full Up, AND Not All Rods Full In and Refuel Platform Over or Near the Core AND Mode Switch in Refuel

A. Correct
All rods are full in therefore a Road block does not occur. The refuel bridge moves unimpeded in this case. A Rod block is initiated due to the bridge being over the core with the mode switch in REFUEL

B. Incorrect
A rod block is initiated with these conditions. Student may choose if they do not understand the rod block circuitry.

C. Incorrect
The refuel bridge is not impeded with these conditions. Student may choose if they do not understand the Refuel Bridge movement protective circuitry.

D. Incorrect
The refuel bridge is not impeded with these conditions. Student may choose if they do

not understand the Refuel Bridge movement protective circuitry. A rod block is initiated with these conditions. Student may choose if they do not understand the rod block circuitry.

Technical Reference(s): SD-281, Rev 9 (page 8 and 9) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2007 NRC ILT

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

10 CFR Part 55 Content:	55.41	2-9
	55.45	7

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.23	
	Importance Rating		4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)

SRO Question 19

Following a rise in condenser backpressure, reactor power has been lowered using recirculation flow. The OATC has determined that load line will exceed 100.64%.

Which of the following is required?

The CRS will direct ________.

A. recirculation flow lowered to stay below the load line limit. A reactivity plan is required

B. recirculation flow lowered to stay below the load line limit. A reactivity plan is **NOT** required

C. control rods inserted to preclude an inadvertent violation of the load line limit. A reactivity plan is required

D. control rods inserted to preclude an inadvertent violation of the load line limit. A reactivity plan is **NOT** required

Proposed Answer: D

Explanation: IPOI-3 Section 5.0 Lowering Power to 35%, step 6 Continuous Recheck Statement If inadvertent entry into the buffer or exclusion areas of the power to flow map, Then comply with the requirements of AOP-255.2 Power/Reactivity Abnormal Change. This will direct control rod insertion, no reactivity plan is required.
AOP-255.2 step 8 follow up actions In the event of inadvertent entry into the area above the power to flow map (ie exceeding the 8 hour average load line of 100.64%) exit this area by inserting control rods
If unable to reduce power below the MELLA limit within one hour, manually scram the reactor

A. Incorrect
This would cause Load Line to increase further. While this may lower power the power to flow conditions would worsen.

B. Incorrect
This would cause Load Line to increase further. While this may lower power the power to flow conditions would worsen.

C. Incorrect, a reactivity plan is not required because the direction is provided with the AOP for the given conditions. Student may select if they believe a reactivity plan is required for all power changes.

D. Correct: Control rod insertion is the only viable method for correcting this condition. This action will reduce the Load Line giving more operating margin to any limits. Direction is provided within the AOP to perform this action. A reactivity plan is not required.

Technical Reference(s):	IPOI-3 Rev 160 P&L 10 and 11 (page 7 of 34) AOP-255.2 rev 48 step 8 (page 6 of 13).	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2013 NRC

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

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.1.43	
	Importance Rating		4.3

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (CFR: 41.10 / 43.6 / 45.6)

Proposed Question: SRO Question 20

The plant is operating at 100% reactor power when the "B" MSR Drain tank dump valve CV1077A fails OPEN. Reactor Power rises ~ 6 MWth and stabilizes.

Based upon these indications what procedure(s) should be entered and what action should be directed to address these conditions?

A. AOP 255.2 Power Reactivity Abnormal **and** AOP 646 Loss of Feedwater Heating. Perform a fast power reduction to 39 MLBM/hr to maintain within Modified Exhaust Pressure Limit

B. AOP 255.2 Power Reactivity Abnormal **and** AOP 646 Loss of Feedwater Heating. Lower recirc flow as necessary to maintain reactor power within licensed power limitations

C. AOP 255.2 **and** IPOI 5 SCRAM, remove the unit from service due to entering unanalyzed loss of feedwater heating region

D. SCRAM the Reactor in accordance with IPOI 5. CLOSE the MSIVs to stabilize RPV pressure

Proposed Answer: B

Explanation: AOP 646 Rev 25. Immediate Action is If reactor thermal power exceeds 1912 MWTh, Then lower reactor power with recirc as necessary to maintain core thermal power less than 1912 MWTh.
AOP 255.2 Power Reactivity Abnormal Rev 48. Step 1 is Take any necessary steps to bring the reactor/reactivity transient under control, including, but not limited to

- Adjusting recirculation flow as necessary to bring power within specified operating limits.
- Scramming the reactor
- Inserting control rods per the current rod pull sheet
- Assuming manual control of a malfunctioning system

A. Incorrect
Fast power reduction is not required. This action is too severe for the given condition. A normal reactivity adjustment is warranted to lower power within the license thermal

limit. In addition the action is not performed to maintain below the MEPL.

B. Correct
Restore power to <1912 MWth with recirc flow reduction. This is an immediate action with AOP 646.

C. Incorrect
Prompt action is taken to reduce power to <1912 MWth. The required action is to lower power to maintain within the Licensed power limit. There is no guidance to shutdown the reactor in this case.

D. Incorrect
Scram is not required, closing the MSIVs is not required. Student may select this action to isolate the MSR from the Main Steam system.

Technical Reference(s): AOP-255.2 Rev 48
AOP 646 Rev 25 (page 2) (Attach if not previously provided)

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 10

55.43 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.25	
	Importance Rating		4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)

Proposed Question: SRO Question 21

A reactor startup is in progress following a refueling outage. Control rod 30-19 was declared inoperable and inserted to position 00. The control room supervisor has directed control rod 30-19 be disarmed.

Which of the following correctly describes how this task can be accomplished?

Per the TS 3.1.3 Bases, Control rod drive 30-19 can be electrically disarmed by removing ___(1)____ or it can be hydraulically disarmed by ___(2)____

A. (1) amphenol connectors from all four directional control valve solenoids for HCU 30-19
(2) closing drive water, exhaust water, and cooling water isolation valves for HCU 30-19

B. (1) fuses at the SCRAM SOLENOID FUSE PANEL for SV-1856 and SV-1855 HCU SCRAM PILOT SOLENOID VALVES for HCU 30-19
(2) closing the drive water, exhaust water, cooling water isolation for HCU 30-19

C. (1) amphenol connectors from all four directional control valve solenoids for HCU 30-19
(2) closing drive water and exhaust water isolation valves for HCU 30-19

D. (1) fuses at the SCRAM SOLENOID FUSE PANEL for SV-1856 and SV-1855 HCU SCRAM PILOT SOLENOID VALVES for HCU 30-19
(2) closing the drive water, exhaust water, isolation for HCU 30-19

Proposed Answer: C

In accordance with TS Bases 3.1.3: (C.1 and C.2) The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. (A.1, A.2, A.3, and A.4) The control rod isolation method should also ensure cooling water to the CRD is maintained.

A. Incorrect: Closing the cooling water isolation to the HCU would cause the Drive to heat up and could result in seal degradation if not corrected. Student may select if they think complete isolation of the CRDM is required.

B. Incorrect: Fuses are not required to be pulled for electrical isolation of the CRDM. This action would fail open the SCRAM inlet and outlet valves. The amphenols for the Directional Control Valves are required to be isolated. Closing the cooling water

isolation to the HCU would cause the Drive to heat up and could result in seal degradation if not corrected. Student may select if they think complete isolation of the CRDM is required.

C. Correct: Per TS Bases disconnecting the amphenols and isolating the Drive hydraulically is a means of disarming the CRD.

D. Incorrect: Fuses are not required to be pulled for electrical isolation of the CRDM. This action would fail open the SCRAM inlet and outlet valves. The amphenols for the Directional Control Valves are required to be isolated.

Technical Reference(s):	Tech Spec 3.1.3 Bases page (B3.1-18 and 19) Amendment 223	(Attach if not previously provided)

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: Monticello

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

10 CFR Part 55 Content:	55.41	5
	55.43	2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.2.43	
	Importance Rating		3.3

Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 22

The plant is shutdown for refueling.

Which one of the following processes is used to track an annunciator window that is in alarm/inoperable for surveillance testing?

A ________________________________ is placed on the annunciator window.

A. white magnetic border

B. green temporary outage lens

C. degraded instrument sticker

D. Maintenance In Progress tag

Proposed Answer: A

Explanation: ACP 1410.1 Operations Working Standards Rev 109.
Section 3.5.5 Instrumentation and Control / Miscellaneous Information
Step 8 Temporary tools (magnets, testing/maintenance white borders, etc) can be used to enhance Operator awareness or to identify pieces of equipment undergoing testing or maintenance.
ODI-14 Temporary Annunciator Identification
Normal outage annunciator: an annunciator window for a system or component which is out of service at least 24 hours due to the normal course of outage work.
Temporary Lense: A shaded plastic that is placed over the annunciator window of a normal outage annunciator
Temporary Annunciator Border: A white plastic frame with a magnetic backing which fits around one annunciator window and is engraved with the words "TESTING/MAINTENANCE across the bottom.
OP-AA-101-1000 Clearance and Tagging
MIP:Tag (Maintenance in Progress) It is used as a flagging device for handswitches, indications, etc.where the device position is not being maintained in a prescribed position.
A MIP tag is for informational purposes only, examples include:

- Ongoing maintenance on a component that is otherwise not tagged such as oil samples, vibration analysis.
- Flagging a component handswitch in the Control Room to indicate that there is ongoing

maintenance / component that is tagged out. The use of a MIP tag is only appropriate when the component (handswitch, indication, etc) is not being maintained in a specified position by the clearance.

- ACP-1408.7 Control of Permanent Plant Instrumentation
- DI is applied to Inoperable or degraded instruments, disable annunciators. Inoperable or disable indication lights

A. Correct

B. Incorrect
Not the purpose of the outage annunciator lens. The student may choose this however this process is used to track annunciators during outages as a result of plant conditions.

C. Incorrect
This is Not the purpose of a DI tag. The student may choose since the annunciator is degraded however another process tracks the conditions in the stem.

D. Incorrect
Not the purpose of a MIP Tag. While there is testing on the equipment which could be construed as maintenance, this tag is not affixed to identify annunciators. Another process tracks the conditions in the stem.

Technical Reference(s): ACP 1410.1 Operations Working Standards Rev 109
ODI-14 Temporary Annunciator Identification
OP-AA-101-1000 Clearance and Tagging Rev 24
ACP-1408.7 Rev 24 Control of Permanent Plant Instrumentation
(Attach if not previously provided)

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	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.3.11	
	Importance Rating		4.3

Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

Proposed Question: SRO Question 23

The plant is shutdown in response to an event. While attempting to close the MSIV's due to elevated Main Steam Line radiation levels, the "B" MSL inboard and outboard MSIVs failed to CLOSE. The following conditions exist:

- OFFGAS stack KAMAN readings - **HI** and stable
- Reactor Building ventilation isolated due to **HI-HI** radiation levels
- SBGT is in operation
- Turbine Building KAMAN reading 3.5e-02 µci/cc and rising
- Turbine Building HVAC has shutdown
- Refuel floor radiation levels are reading 100 mr/hr and rising slowly
- CRD rebuild room radiation levels are reading 1000mr/hr and rising slowly

What procedure and action should be directed to address the conditions above?

A. EOP 3 SECONDARY CONTAINMENT CONTROL, Restart the Reactor Building ventilation systems by installing Defeat 9

B. EOP 3 SECONDARY CONTAINMENT CONTROL, perform an Emergency Depressurization based upon 2 areas above max safe.

C. EOP 4, perform an Emergency Depressurization based upon exceeding Site Area Emergency Radiation Release levels

D. EOP 4, Stop the Main Plant Exhaust Fans, Stop TB Supply Fans, verify at least 1 TB Exhaust Fan operating at High Speed

Proposed Answer: D

Explanation:

A. Incorrect
EOP-3 CRS states if all the following conditions apply
RB Vent shaft RIM-7606A(B) is below 8 mR/hr. This is not true. Defeat 9 is not allowed.
Student may select if they do not understand the limitation of use of the defeat.

B. Incorrect:
EOP-3 SC-4 decision will RPV pressure reduction decrease leakage into secondary

containment. Stem does not indicate that a primary system leak is present in secondary containment. SC-4 would be answered No and continue to step SC-8 shutdown per IPOI-3,4, or 5.

C. Incorrect:
TB radiation levels are at the Alert, conditions are not met to ED per EOP-4. Student may select if they do not determine the appropriate threshold to ED.

D. Correct
EOP-4 CRS provides the guidance to perform these actions.

Technical Reference(s): EOP-3 Rev 22
EOP-4 Rev 22
EAL-01 Rev 11
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL Board EAL-01
EOP-3 Table 6

-

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 11
55.43 4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.32	
	Importance Rating		4.0

Knowledge of operator response to loss of all annunciators.(CFR: 41.10 / 43.5 / 45.13)

Proposed Question: SRO Question 24

The plant is operating at 100% reactor power when the following conditions are observed:

- All control rods scram full in
- All systems respond as designed to the scram and are indicating properly
- No annunciators from 1C08, 1C05, 1C04, or 1C03 alarm.

What (if any) EAL is required to be declared?

A. NO EAL is required

B. Unusual Event

C. Alert

D. Site Area Emergency

Proposed Answer: C

Explanation: EAL SA4.1 Unplanned Loss of approximately 75% or more of ANY of the following for 15 minutes or more:

- 1C03, 1C04, and 1C05 annunciators
- 1C03, 1C04, and 1C05 indications
- Radiation monitor indications

AND
Either of the following

- A SIGNIFICANT Transient is in progress
- Compensatory indications are unavailable

EAL Bases Document EBD S (system malfunction) describes a significant transient includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

A. Incorrect
AN EAL threshold has been met. Student may select if they do not believe there are enough annunicators lost to declare.

B. Incorrect

An Alert should be declared for the given conditions. If the student does not determine a significant transient in progress they may select this choice.

C. Correct
Alert threshold has been met

D. Incorrect
Alert threshold has been met
Site threshold would be met if Compensatory indications were not unavailable. Student may select if they believe compensatory indications are not available either.

Technical Reference(s):	AOP-302.2 LOSS OF ALARM PANEL POWER EAL-01 Rev 11 EAL Bases Document EBD S Rev 10 (page 17 of 31) describes significant transient.	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL Board EAL-01

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G2.4.50	
	Importance Rating		4.0

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

Proposed Question: SRO Question 25

The plant was operating at 100% reactor power with **Torus Cooling in service**.

- Day 1 of 7 per TS 3.5.1.B

The following annunciator is received:

- 1C03A(A-8), "A" Core Spray System AUTO Initiation

Thirty (30) seconds later the following annunciator is received:

- 1C03A(A-9), "A" Core Spray Pump 1P-211A Trip or Motor Overload

What is the **FIRST** required action for this condition?

A. Restore Drywell Spray Subsystem to FUNCTIONAL within 8 hours

B. Restore the RHR Subsystem to OPERABLE within 72 hours

C. Restore the "A" Core Spray Subsystem to OPERABLE within 7 days

D. Enter LCO 3.0.3 IMMEDIATELY

Proposed Answer: D

Explanation: Tech Specification 3.5.1 B One Low Pressure ECCS subsystem inoperable for reasons other than Condition A (One RHR Pump inoperable).
3.5.1.N Two or more low pressure ECCS subsystems inoperable for reasons other than Conditions C or D (Enter LCO 3.0.3)
Condition C One CS subsystem inoperable and one or two RHR pumps inoperable (RHR pumps are not inoperable in this situation, the LPCI mode is inoperable due to being in Torus Cooling.
Condition D Both CS subsystems inoperable. Only the "A" CS subsystem is inoperable.
OI-149 Section 5.4 Normal Torus Cooling Continuous Recheck Statement
If Torus Cooling is operating when LPCI is required to be Operable Then LPCI shall be declared inoperable and the Technical Specification for ECCS-Operating and RPV Water Inventory Control complied with

A. Incorrect:
Drywell Spray remains operable in these conditions. Student may incorrectly assume the LPCI inoperability due to Torus Cooling also makes Drywell Spray inoperable.

B. Incorrect:
3.5.1.B was entered for the LPCI subsystem inop while in Torus cooling. Condition C is not applicable for the given conditions.

C. Incorrect:
Although Core Spray is inoperable in the given condition, student must realize that two low pressure ECCS subsystems are inoperable and correct condition would be N enter LCO 3.0.3

D. Correct:

Technical Reference(s): TS 3.5.1 rev 262
OI-149 rev 169 Section 5.4 Normal Torus Cooling (Attach if not previously provided)

Proposed References to be provided to applicants during examination: LCO 3.5.1 and TRM 3.5.1

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43 5