Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295030	EA2.01
	Importance Rating		4.2

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

Proposed Question: SRO Question # 76

During an accident the following plant conditions exist:

- RPV pressure is 440 psig and lowering
- RPV water level is 40 inches and lowering
- Drywell pressure is 4 psig and rising
- Drywell temperature is 160°F and rising
- Torus water level is 8.9 ft and stable
- Torus pressure is 3.9 psig and rising
- All RHR pumps are running
- HPCI and RCIC are injecting
- All Control Rods are Full In

Which one of the following is required to be directed based upon the above conditions?

A. IAW EOP-1, Enter EOP-ED and Emergency Depressurize using the ADS SRVs.

B. IAW EOP-2, Primary Containment Control, initiate Drywell Spray.

C. IAW OI 151, Core Spray, attempt to raise Torus level.

D. IAW EOP-1, Anticipate ED and open Bypass Valves.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – RPV level has not yet lowered per the EOPs to require ED. No other parameters dictate performing an ED.

B. Incorrect – With RPV level lowering, initiating Torus Spray is not performed unless adequate core cooling is assured.

C. Correct – IAW EOP 2 step T/L-1 with Torus level < 10.1 " then attempt to raise level using HPCI or Core Spray. (HPCI is currently injecting so Core Spray may be used

D. Incorrect – Anticipating ED is not permitted with an RPV level issue.

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

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001	AA2.06
	Importance Rating		3.3

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation

Proposed Question: SRO Question # 77

The plant is operating in MODE 1 at 48% power with the following conditions:

- Core Flow is 27 Mlbm/hr
- Both Recirculation Pumps are at 46% speed
- The “A” Recirculation MG Lube Oil Pressure lowers to 28 psig due to a leak

15 seconds later:
(1) What is the status of the “A” Recirc Loop Flow indication? AND
(2) What actions are required to be directed?

A. (1) NOT accurate

(2) IAW OI 264, Reactor Recirc System, direct the M/G Set Emergency DC Lube Oil Pump supply breaker 1D4202 be turned ON, and start the pump to raise lube oil pressure.

B. (1) accurate

(2) IAW OI 264, Reactor Recirc System, direct the M/G Set Emergency DC Lube Oil Pump supply breaker 1D4202 be turned ON, and start the pump to raise lube oil pressure.

C. (1) NOT accurate

(2) IAW AOP-255.2, Power/Reactivity Abnormal Change, and evaluate Power to Flow, if in the Exclusion region and no instabilities exist, direct RO to insert Control Rods or raise Recirc Flow to exit the Region.

D. (1) accurate

(2) IAW AOP-255.2, Power/Reactivity Abnormal Change, and evaluate Power to Flow, if in the Exclusion region and no instabilities exist, direct RO to insert Control Rods or raise Recirc Flow to exit the Region.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – [Part 1 is correct, part 2 is incorrect] The A RR pump would trip due to low RRMG oil pressure, but with “A” RR MG Set <50% speed (46%), there will be forward flow through both Loops. With the “A” RR MG Breakers OPEN, flow subtraction will cause Total Core Flow Indication to be inaccurate. During a RR pump trip, AOP 264 is entered, not the OI, and there is no direction in the procedure to start the DC pump, it should auto start as lube oil pressure continues to lower.

B. Incorrect – [Part 1 is incorrect, part 2 is incorrect] The A RR pump would trip due to low RRMG oil pressure, but with “A” RR MG Set <50% speed (46%), there will be forward flow through both Loops. With the “A” RR MG Breakers OPEN, flow subtraction will cause Total Core Flow Indication to be inaccurate. During a RR pump trip AOP 264 is entered, not the OI, and there is no direction in the procedure to start the DC pump, it should auto start as lube oil pressure continues to lower.

C. Correct – IAW SD 264 With Lube Oil less than 30 psig for 6 seconds will cause an MG TRIP. With A RR MG Set <50% speed (46%), there will be forward flow through both Loop. With the A RR MG Breakers OPEN, Flow subtraction will cause Total Core Flow Indication to be inaccurate. It is required to raise B Recirc MG Set Speed or Insert Control Rods to exit the Exclusion Zone per AOP-255.2, Power / Reactivity Abnormal Change.

D. Incorrect – [Part 1 is incorrect, part 2 is correct] The A RR pump would trip due to low RRMG oil pressure, but with “A” RR MG Set <50% speed (46%), there will be forward flow through both Loops. With the “A” RR MG Breakers OPEN, flow subtraction will cause Total Core Flow Indication to be inaccurate. . It is required to raise B Recirc MG Set Speed or Insert Control Rods to exit the Exclusion Zone per AOP-255.2, Power / Reactivity Abnormal Change.

Technical Reference(s): AOP 255.2 Rev .35 page 5 Caution
SD 264 Rev.12 page 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 4054

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6/10/11, focused question more on K/A statement, removed pump tripping options, but pump tripping still has to be understood to correctly answer the question.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	295018	AA2.01	
	Importance Rating			3.4

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Component temperatures

Proposed Question: SRO Question # 78

The plant is operating in MODE 1 at 100% power with the following conditions:

- GSW temperatures have started to rise
- Field operators confirm that return line temperatures for GSW loads are rising

(1) What are loads directly affected by GSW?
(2) Actions to be directed?

A. (1) Instrument and Service Air Compressors, Drywell Cooling Coils and the Isophase Bus Duct Cooler.
(2) Enter into AOP 411, GSW Abnormal Operation, and direct the reducing of Main Generator load to less than 11,000 amps
AND
Direct the operator to take manual control of TC-4717, GSW FROM GENERATOR HYDROGEN COOLERS and throttle OPEN the valve.

B. (1) Steam Tunnel Cooler, Exciter Air Cooler and the Isophase Bus Duct Cooler.
(2) Enter into AOP 411, GSW Abnormal Operation, and direct the reducing of Main Generator load to less than 11,000 amps
AND
Direct the operator to take manual control of TC-4717, GSW FROM GENERATOR HYDROGEN COOLERS and throttle OPEN the valve.

C. (1) Control Building Chiller, Drywell Cooling Coils and the Steam Tunnel Cooler.
(2) Enter AOP 573, Primary Containment Control, and direct venting the drywell to maintain 1.0 to 1.5 psig
AND
Enter IPOI 4, Shutdown, and direct a Fast Power Reduction.

D. (1) Steam Tunnel Cooler, Exciter Air Cooler and the Isophase Bus Duct Cooler.
(2) Enter AOP 573, Primary Containment Control, and direct venting the drywell to maintain 1.0 to 1.5 psig
AND
Enter IPOI 4, Shutdown, and direct a Fast Power Reduction.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Drywell Cooling Coils are Well Water loads

B. Correct – These are loads and actions referenced in AOP 411

C. Incorrect – Drywell Cooling Coils are Well Water Loads and Venting the Drywell is not require in a loss of GSW.

D. Incorrect – Venting the Drywell is not require in a loss of GSW

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

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11- NRC OK with change

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025	2.4.45
	Importance Rating		4.3

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm. (High Reactor Pressure)

Proposed Question: SRO Question # 79

The plant is operating in MODE 1 at 100% power with the following conditions:

- Annunciator 1C05B (D-4), REACTOR VESSEL HI PRESSURE ALARM alarms
- Analysis of PR-4563/4564, REACTOR PRESSURE/REACTOR WATER LEVEL indicates that reactor pressure has been slowly rising over the shift until the alarm setpoint was reached.

Which one of the following actions is the SRO required to direct?

A. Enter AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure and within 15 minutes lower reactor pressure below the Technical Specification Limit.

B. Enter AOP-255.2, Power/Reactivity Abnormal Change, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure and within 15 minutes lower reactor pressure below the Technical Specification Limit.

C. Enter AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.1.2 Reactivity Anomalies and within 15 minutes lower reactor power below 1912 Mwth by lowering reactor pressure.

D. Enter AOP-255.2, Power/Reactivity Abnormal Change and Technical Specifications 3.1.2 Reactivity Anomalies and within 15 minutes lower reactor power below 1912 Mwth by lowering reactor pressure.

Proposed Answer: A

Explanation (Optional):

A. Correct – Annunciator 1C05B, REACTOR VESSEL HI PRESSURE ALARM is an entry condition for AOP-262, Loss of Reactor Pressure Control, and Technical Specifications 3.4.10, Reactor Steam Dome Pressure. The AOP directs lowering reactor pressure and Technical Specifications requires lowering reactor pressure below 1025 psig within 15 minutes.

B. Incorrect – There are no entry conditions for AOP-255.2, Power/Reactivity Abnormal Change. Although reactor power will rise OI-693.1, states DAEC experience shows that core thermal power may rise approximately two-thirds of one megawatt for each psig of pressure rise. Assuming a pressure rise of 25 psig (from 1015 to 1040) would only cause a power rise of ~17 Mwth, which is below the 3% power (~57 Mwth) rise needed to cause and APRM High Alarm.

C. Incorrect - Lowering power below 1912 Mwth may not lower pressure below 1025 psig which the Tech Specs require.

D. Incorrect There are no entry conditions for AOP-255.2, Power/Reactivity Abnormal Change. Although reactor power will rise OI-693.1, states DAEC experience shows that core thermal power may rise approximately two-thirds of one megawatt for each psig of pressure rise. Assuming a pressure rise of 25 psig (from 1015 to 1040) would only cause a power rise of ~17 Mwth, which is below the 3% power (~57 Mwth) rise needed to cause and APRM High Alarm. Additionally lower power below 1912 Mwth may not lower pressure below 1025 psig which the Tech Specs require.

Technical Reference(s): AOP 262, Tech Specs 3.4.10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019	2.4.35
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects. (Partial or Total Loss of Inst. Air)

Proposed Question: SRO Question # 80

The plant is in MODE 2 with reactor pressure at 120 psig with the following conditions:

- Steam seals are in service
- Condenser backpressure at 5.5” Hg absolute
- 1K1 is tagged out for maintenance

Then an instrument Air leak develops in the heater bay and the plant conditions are now as follows:

- 1K-90 A and B are running and loaded
- 1K-90C did not start, maintenance is working on it
- Instrument and Service Air pressure is 86 psig and slowly lowering

Which of the following:
(1) Identify the procedure that is required to be entered?
(2) Identify an action that must be directed to the NSPEO?

A. (1) Enter AOP 518, Failure Of Instrument And Service Air
(2) Direct the NSPEO to place the Instrument Air Dryers IT-265A/B in FAIL SAFE mode by placing HS-3046A/B in the ON position

B. (1) Enter AOP 518, Failure Of Instrument And Service Air
(2) Direct the NSPEO to place the Instrument Air Dryers IT-265A/B in FAIL SAFE mode by placing HS-3046A/B in the OFF position

C. (1) Enter IPOI-5, Reactor Scram
(2) Direct the NSPEO to throttle OPEN MO-4165, Offgas Jet Compressor Regulator Bypass, to raise steam flow to at least 4300 lbm/hr.

D. (1) Enter IPOI-5, Reactor Scram
(2) Direct the NSPEO to isolate Deluge 11 (Aux Transformer) by shutting V-33-73, East Turbine Building.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – AOP 518 would be entered for these plant conditions. The Instrument Air Dryers are placed in the Fail Safe mode by placing HS-3046A/B in OFF.

B. Correct - AOP 518 would be entered for these plant conditions. The Instrument Air Dryers are placed in the Fail Safe mode by placing HS-3046A/B in OFF.

C. Incorrect – IPOI-5 is not REQUIRED to be entered at an air pressure of 86 psig and because the turbine is not in operation with only 120 psig reactor pressure the low condenser vacuum does NOT require a scram. At the given plant conditions, the Offgas System is not in service.

D. Incorrect – IPOI-5 is not REQUIRED to be entered at an air pressure of 86 psig and because the turbine is not in operation with only 120 psig reactor pressure the low condenser vacuum does NOT require a scram. Isolate Deluge 11 is an action at 50 psig Instrument Air pressure, or if pressure is lowering rapidly.

Technical Reference(s): AOP 518 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11- NRC OK with enhancement

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023	2.2.22
	Importance Rating		4.7

Equipment Control: Knowledge of limiting conditions for operations and safety limits. (Refueling Accidents)

Proposed Question: SRO Question # 81

The plant is in MODE 5 with the fuel pool gates installed for a RFO with the following conditions:

- Fuel is being moved in the Spent Fuel Pool.

Then, annunciator FUEL POOL COOLING PANEL 1C-65/1C-66 TROUBLE 1C04B D-2 alarms. The following plant conditions exist:

- The cause of the alarm is Skimmer Surge Tank Low Level
- Spent Fuel Pool level is 35 feet and slowly lowering
- Fuel Pool Cooling Pump 1P-214A is running
- Refuel Floor ARMs have increased by 2 mr/hr

(1) Has the TS LCO for Spent Fuel Pool Level requirements been exceeded, if so, what actions are required?
(2) What, if any, EAL is required to be declared?

A. (1) The TS LCO is met.
(2) NO EAL has been exceeded.

B. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended immediately.
(2) An Unusual Event must be declared.

C. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended within 1 hour if level is not corrected.
(2) NO EAL has been exceeded.

D. (1) The TS LCO has been exceeded and movement of irradiated fuel assemblies in the spent fuel storage pool must be suspended within 4 hours if level is not corrected.
(2) An ALERT must be declared.

Proposed Answer: B

Explanation (Optional):

A. Incorrect –The LCO has been exceeded and UE is required IAW EAL-02, RU2.1, Unplanned valid Refuel Floor ARM reading increase with an uncontrolled loss of reactor cavity, fuel pool, or fuel transfer canal water level with all irradiated fuel assemblies remaining covered by water as indicated by any of the following:
- Report to control room
- Valid fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering
- Valid WR GEMAC Floodup indication (LI-4541) coming on scale

B. Correct – IAW ARP 1C04B A-4 Section 3.7 & 3.8. TS 3.7.8 LCO limit is 36 feet, action is required to immediately suspend fuel movement. An Unusual Event must be declared because fuel pool level indication (LI-3413) LESS THAN 36 feet and lowering.

C. Incorrect - A UE must be declared

D. Incorrect – An ALERT level has not been exceeded

Technical Reference(s): TS 3.7.8
EAL-02 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL-01,02

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

55.43 2, 7

Facility operating limitations in the technical specifications and their bases.
Fuel handling facilities and procedures.

Comments:
6-10-11-NRC OK- Not unsat - SAT

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295021	2.4.9
	Importance Rating		4.2

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (Loss of Shutdown Cooling)

Proposed Question: SRO Question # 82

The plant is in MODE 4, with the following:

- The plant has been operating shutdown cooling for 36 hours.
- "A" Recirculation Pump is in operation.
- An equipment failure results in the loss of both loops of Shutdown Cooling.

(1) Which one of the following actions is required by plant Technical Specifications?
(2) What is the NRC Reportability requirement, if any?

A. (1) Restore one loop of shutdown cooling to operation within one hour
(2) This is reportable to the NRC within four hours.

B. (1) Restore both shutdown cooling subsystems to operation within one hour
(2) This is NOT reportable to the NRC.

C. (1) Verify an alternate method of decay heat removal is available within one hour.
(2) This is reportable to the NRC within eight hours.

D. (1) Verify an alternate method of decay heat removal is available immediately.
(2) This is NOT reportable to the NRC.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – {part 1 is correct, part 2 is incorrect] Restoring one loop of SDC to operation within one hour is a correct TS action, but the NRC Reportability is within eight hours.

B. Incorrect – [part 1 is correct, part 2 is incorrect] The requirement is to restore the availability of Decay Heat Removal within one hour (by administrative means), but this event is reportable.

C. Correct – .It is required to restore the availability of decay heat removal within one hour, and ACP 1410.3 requires a eight hour NRC notifications IAW 50.72(b)(3)(v)(B).

D. Incorrect – [part 1 incorrect, part 2 incorrect] Immediate action is not required for restoring Decay Heat Removal, the TS gives one hour. This is also a reportable event.

Technical Reference(s): TS 3.4.8 and 3.4.8 Bases
ACP 1402.3
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: ACP 1402.3, Attachment 3

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

55.43 2,5

Facility operating limitations in the technical specifications and their bases.
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11-Revised to add Reportability aspect of this event, added table 3 of the Reportability ACP to references, deleted the TS reference (students will not be given this ts)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	295020	AA2.04
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor pressure

Proposed Question: SRO Question # 83

The plant was operating in MODE 1 at 100% power when an electrical transient occurred resulting in an automatic reactor scram.

Following the scram:

- All Control Rods are Full In
- Reactor level lowered to 155 inches and recovered to 178 inches and is slowly rising
- Drywell Pressure is 1.7 psig and stable
- Drywell temperature is 120°F and stable
- Torus Water Level is 10.2' and rising slowly
- The Main Turbine is tripped
- The SBDGs are running unloaded with the Essential Buses on their normal supply

Then, another electrical transient occurs resulting in a total loss of RPS power. RPS power cannot be restored.

Which of the following is a successful pressure control strategy that you would select when directing EOP 1, RPV Control?

A. Cool down the RPV with the Main Turbine Bypass Valves, not exceeding 100°F/hr cooldown rate.

B. Place all SRV handswitches in AUTO and bypass Main Condenser High Backpressure Isolation by installing Defeat 17 to open the MSL drain valves, MO-4423 and MO-4424.

C. Place RWCU in the Recirc Mode of operation IAW SEP 302.1 OR the Drain Mode of operation IAW SEP 302.2.

D. Place RCIC or HPCI in pressure control mode. EOP Defeat 1 may be installed

Proposed Answer: D

Explanation (Optional):

A. Incorrect – With the MSIVs closed, the bypass valves are not available

B. Incorrect – The MSIVs are already closed so installing this Defeat would not be useful

C. Incorrect – With total loss of RPS, PCIS group 5 would not allow RWCU to be used.

D. Correct – HPCI and RCIC are available for use, and Defeat 1 is authorized

Technical Reference(s): EOP-1 Step RC/P-4 and Table 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	295017	2.1.19
	Importance Rating		3.8

Conduct of Operations: Ability to use plant computers to evaluate system or component status. (High Offsite Release rate) **KA Match Justification** – Offsite release rate is determined by dose assessment using actual meteorology, field surveys and by using the KAMAN monitors. This question involves evaluating equipment status based on KAMAN indications

Proposed Question: SRO Question # 84

The plant was operating in MODE 1 at 100% power with the following conditions:

- An unisolable coolant leak occurred in the Reactor Building
- RPV level lowered to the point that fuel became uncovered and fuel damage occurred
- RPV level was recovered to normal
- Offgas Kaman readings are 6.3E+0 µCi/cc and slowly rising
- The RED annunciator on panel 1C35 for REACTOR BLDG KAMAN 3, 4, 5 ,6 , 7,& 8 HI RAD OR MONITOR TROUBLE alarms
- The Reactor Building Exhaust Fans (1V-EF-11A & B) and the Main Plant Exhaust Fans (1V-EF-1, 2, & 3) responded as designed

Given the above:
(1) What actions must be directed to mitigate this condition?
(2) What is the highest EAL that must be declared?

A. (1) Enter EOP-4, Radioactivity Release Control, and direct operators to TRIP the Main Plant Exhaust Fans.
(2) Declare an ALERT when it is determined the release will last for greater than 15 minutes.

B. (1) Enter EOP-4, Radioactivity Release Control, and direct operators to RESTART the Main Plant Exhaust Fans.
(2) Declare an ALERT when it is determined the release will last for greater than 60 minutes.

C. (1) Enter OI-734, Reactor Building HVAC, and direct operators to TRIP the Reactor Building Exhaust Fans
(2) Notice of Unusual Event (NOUE) when it is determined the release will last for greater than 60 minutes.

D. (1) Enter OI-734, Reactor Building HVAC, and direct operators to RESTART the Reactor Building Exhaust Fans.
(2) Site Area Emergency when it is determined the release will last for greater than 15 minutes.

Proposed Answer: A

Explanation (Optional):

A. Correct – At <170 inches a Group 3 isolation occurs which trips EF-11A&B, closes AD-13A & B, and aligns SBGT to draw on the RB Vent Shaft. EF1/2/3 continue to run and draw on the Main Plant Exhaust Plenum. The RB Vent Shaft and the MP Exhaust Plenum are physically separated by only a wall which, in the history of the plant, has been found to be cracked. Also the dampers AD-13A/B could be leaking, also allowing the RB Vent Shaft to flow to the MP Exhaust Plenum and out past EF-1/2/3 which normally continue to run after a Group 3 isolation. This is a real enough concern that the is a P&L in the Reactor Building HVAC OI, a Continuous Recheck statement in EOP-4 and Steps in ARP 1C35A C-3. EOP-4 is not provided. The conditions given lead to an EAL of RA 1.2

B. Incorrect - A high concentration of activity will cause this alarm. As part of the EOP-4 Recheck statement, Operators are directed to restart the Turbine Bldg Exhaust Fans, not Main Plant Exhaust Fans which would still be running. The given conditions do lead to an ALERT, however, the time required is only 15 minutes. 60 minutes is used in this EAL block, but not for an ALERT.

C. Incorrect - . The NOUE would be a correct EAL call, but is not the highest EAL call for the given conditions.

D. Incorrect – The LOCA was in the Secondary Containment and operators are directed to restart the Turbine Bldg Exhaust Fans, not Reactor Building Exhaust Fans. SAE would be required to be call, but the given radiation levels are not the SAE setpoint.

Technical Reference(s): OI-734, P&L #4; EOP-4; ARP 1C35A, C-3 ARP 1C05B C-8 EAL-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SRO 6.72.01.06 Evaluate plant conditions and CR indications and determine if any step of EOP-4 should be performed (As available)

Question Source: Bank # DAEC 2005

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Revised question to add EAL choice for C & D.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	295002	AA2.04
	Importance Rating		2.9

Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Offgas system flow

Proposed Question: SRO Question # 85

The plant is in MODE 2 at 6% power and stable with the following plant conditions:

- RPV pressure is 780 psig and stable
- The Mechanical Vacuum pump is OFF
- Annunciator 1C34, C-3 OFFGAS JET COMPRESSOR LO STEAM FLOW alarms
- CV-4151, OG JET COMPRESSOR 1S-111 SUCTION ISOLATION, is partially OPEN and is being controlled using Offgas Flow Controller HIC-4151
- Offgas flow is lowering
- Condenser vacuum is degrading and there is no indication that the loss of Offgas is due to a steam leak

Which one of the following is required?

A. Enter OI 691, Condenser Air Removal, and direct the shutdown of the Condenser Air Removal System.

B. Enter OI 150, Reactor Core Isolation Cooling, and direct an RO to place RCIC on service in the Pressure Control Mode.

C. Enter AOP 672.1, Loss of Offgas System, and direct that MO-4156 be throttled open until FI-4150 indicates at least 4300 lbm/hr

D. Enter AOP 672.1, Loss of Offgas System, and direct that CV-4151 be failed open and control pressure using MO-4151.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – this is the action if a steam leak is the cause.

B. Incorrect – these are actions are required if power is >10% (reactor scram, and RCIC is being used for reactor pressure control).

C. Correct – With CV-4151, OG JET COMPRESSOR 1S-111 SUCTION ISOLATION, partially open and 1C34, C-3 OFFGAS JET COMPRESSOR LO STEAM FLOW in alarm AOP-672.1 immediate action Step 7 directs Throttle open MO-4156 in the Recombiner Room until FI-4150 indicates at least 4300 lbm/hr.

D. Incorrect – IAW AOP 672.1 Steps 10 & 13 CV-4151 would only be failed open if CV-4151 could not be opened, the question stem states the valve is partially open and has no indication it cannot be opened further.

Technical Reference(s): AOP 672.1, Step 7, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11- NRC OK fix explanations. Revised question (combined bullets to ensure only one correct answer).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	218000 A2.03	
	Importance Rating		3.6

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: Loss of air supply to ADS valves: Plant-Specific

Proposed Question: SRO Question # 86

Following a small steam break in the Drywell the following conditions exist:

- Containment pressure is 4 psig, and rising very slowly
- The Crew has entered EOP-1, RPV Control and EOP-2, Primary Containment Control
- SRV manual initiations were required to cooldown the RPV
- All automatic actions occurred as expected

Given the above:
(1) What is the impact of a LONG TERM Nitrogen supply isolation? AND
(2) What actions are required to be directed?

A. (1) SRVs will fail to open for manual RPV pressure control OR for initiation of ADS.
(2) Direct the installation of EOP Defeat 11, Containment N_2 Isolation Defeat, and place CV-4371A GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to ensure SRV Nitrogen Accumulators remain charged.

B. (1) SRVs will still open in response to an RPV high pressure condition OR a manual signal, but will not an open for an ADS initiation.
(2) Direct the installation of EOP Defeat 11, Containment N_2 Isolation Defeat and place CV-4371A GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to directly supply nitrogen to the SRVs.

C. (1) SRVs will fail to open for manual RPV pressure control or for initiation of ADS.
(2) Direct the installation of EOP Defeat 9, Group 3, High DW Press & Rx Low Level Isolation Defeat, and place CV-4371C GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to ensure SRV Nitrogen Accumulators remain charged.

D. (1) SRVs will still open in response to an RPV high pressure condition OR a manual signal, but may not an open for an ADS initiation.
(2) Direct the installation of EOP Defeat 9, Group 3, High DW Press & Rx Low Level Isolation Defeat, and place CV-4371C GROUP 3 OVERRIDE keylock switch S583B in OVERRIDE OPEN to directly supply nitrogen to the SRVs.

Proposed Answer: A

Explanation (Optional):

A. Correct - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. The correct EOP Defeat is #11 which is specifically intended for this condition.

B. Incorrect - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. Without Nitrogen pressure the SRVs cannot be manually opened.

C. Incorrect - EOP Defeat 9 does NOT contain a step for bypassing the Compressor Isolation Valves. On a Group 3 isolation signal (SBGT Lockout), both of these valves will close. CV-4371A has a 2 position keylock NORM-OVERRIDE OPEN switch (Key removable in NORM) which can be used to open the valve with a Group 3 signal present. CV-4371C has no such feature. Defeat 9 is intended for defeating the High Drywell Pressure and RPV Low Water Level Isolations from Group 3 PCIS logic. This enables restoration of Reactor Building ventilation and/or containment vent valves.

D. Incorrect - Both manual RPV pressure control or for initiation of ADS require Nitrogen to open the SRV. Without Nitrogen pressure the SRVs cannot be manually opened. EOP Defeat 9 does NOT contain a step for bypassing the Compressor Isolation Valves. On a Group 3 isolation signal (SBGT Lockout), both of these valves will close. CV-4371A has a 2 position keylock NORM-OVERRIDE OPEN switch (Key removable in NORM) which can be used to open the valve with a Group 3 signal present. CV-4371C has no such feature. Defeat 9 is intended for defeating the High Drywell Pressure and RPV Low Water Level Isolations from Group 3 PCIS logic. This enables restoration of Reactor Building ventilation and/or containment vent valves.

Technical Reference(s): SD-183.1
SD 573, pg 25
EOP Defeat 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11-NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	259002 A2.06	
	Importance Rating		3.4

Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of controller signal output

Proposed Question: SRO Question # 87

The plant is operating in MODE 1 at 50% power with the following conditions:

- The Feedwater Level Control System is in THREE element control
- FT-1581, the "A" Feedwater Flow Transmitter output lowers to zero (0)

(1) What procedure you would enter, and required actions to be directed? AND
(2) What other actions you will direct?

A. (1) OI-644, Feedwater/Condensate, Section 4.3, Selecting Reactor Water Level Control Input, and direct the RO to shift feedwater control to single element.
(2) No other actions necessary, the circuit will correct RPV level transient with no further actions.

B. (1) AOP-644, Feedwater/Condensate Malfunction, direct the RO to take manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.
(2) Direct the RO to throttle the FRV's CLOSED to recover RPV water level, then stabilize the level in the green band.

C. (1) OI-644, Feedwater/Condensate, Section 4.3, Selecting Reactor Water Level Control Input, and direct the RO to shift feedwater control to single element.
(2) Direct the RO to throttle the FRV's OPEN to recover RPV water level, then stabilize the level in the green band.

D. (1) AOP-644, Feedwater/Condensate Malfunction, and direct the RO to take manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.
(2) No other actions necessary, the circuit will correct RPV level transient with no further actions.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - This section of OI-644 does not specify placing feedwater in single element control. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.

B. Correct - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577.

C. Incorrect - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow. This section of OI-644 does not specify placing feedwater in single element control. Abnormal level indication and/or level annunciators will require entry into AOP 644. The AOP directs taking manual control of MASTER FEED REG VALVE AUTO/MANCONTROL LC-4577

D. Incorrect - RPV water level will rise because FWLC will open the Feed Regulating Valves due to steam flow being higher than feed flow.

Technical Reference(s): AOP-644, SD-644 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11-NRC enhanced not unsat. Enhanced question with minor text changes.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	206000	2.4.18
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: SRO Question # 88

The reactor was scrammed due to an un-isolable Feedwater line break outside the Primary Containment. Busses 1A1 and 1A2 tripped and locked out due to the feedwater leak. Current plant conditions are as follows:

- RPV water level is 100 inches, rising slowly
- HPCI and RCIC are injecting
- Drywell pressure is 2.7 psig, stable
- Torus water level is 7.1 feet, lowering at 0.1 feet per minute
- Torus water temperature is 88°F, rising slowly

(1) What procedure is required to be entered and what actions are required to be taken?
(2) What is the bases for this action?

A. (1) Enter RCIC Quick Response Card (QRC) and HPCI QRC, and direct the RO to TRIP HPCI and RCIC
(2) Prevent direct pressurization of the Primary Containment

B. (1) Enter SEP 307, and direct to RO to open turbine Bypass Valves in anticipation of Emergency Depressurization
(2) Minimize heat addition to the torus

C. (1) Enter the Emergency Depressurization EOP and direct the RO to open 4 ADS SRVs to emergency depressurize the RPV
(2) Prevent exceeding the HCTL curve

D. (1) Enter the Emergency Depressurization EOP and direct the RO to open 4 ADS SRVs to emergency depressurize the RPV
(2) Due to uncovering the downcomers

Proposed Answer: D

Explanation (Optional):

A. Incorrect – HPCI is required to be secured at 5.8 ft torus level, not 7.1 ft

B. Incorrect – EOP 2 requires ED BEFORE torus level reaches 7.1 ft

C. Incorrect - Torus level of 7.1 ft corresponds to a loss of PSP of the containment, not exceeding the HCTL curve.

D. Correct - A torus level of 7.1 ft. corresponds to the bottom of the drywell-to-torus downcomers. Torus levels below 7.1 ft. would result in loss of the pressure suppression function of the primary containment (e.g., during a LOCA, steam entering the torus would not be fully condensed).

Technical Reference(s): EOP-2 Bases, pg 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	262001 2.4.50	
	Importance Rating		4.0

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (AC Distribution)

Proposed Question: SRO Question # 89

The plant is operating in MODE 1 at 100% power with the following conditions:

- Several annunciators alarm on the 1C08 panel
- Fifteen minutes later the following annunciators are still in their alarm condition:
  - 1C08A, A-5, BUS 1A3 LOCKOUT TRIP
  - 1C08B, A-6, BUS 1A4 LOCKOUT TRIP
  - 1C08B, A-3, S/U XFMR TO 1A1 BREAKER 1A102 TRIP
  - 1C08A, A-8, S/U XFMR TO 1A2 BREAKER 1A202 TRIP

As the CRS you must direct the crew to verify the…

A. Standby Diesel Generators are shutdown and declare a SITE AREA EMERGENCY based on a loss of all offsite power and loss of all onsite AC power to Essential Buses for 15 minutes.

B. Standby Diesel Generators are shutdown and declare an ALERT based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

C. Standby Diesel Generators are supplying Buses 1A3 and 1A4 and declare a SITE AREA EMERGENCY based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

D. Standby Transformer is supplying Buses 1A2, and either 1A3 or 1A4 and declare a ALERT based on AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Proposed Answer: A

Explanation (Optional):

A. Correct – With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses. The SBDG will start but will not have cooling water and the ARPs for the respective lockout annunciators direct shutting down the SBDGs. With Breakers 1A101 and 1A201 tripped no power is available to buses 1A1 and 1A2 from either the Startup or Standby Transformers. Because the plant tripped on a loss of RPS there is no power available from the Auxiliary Transformer. Therefore the station is in a Blackout condition and has been for 15 minutes. IAW EAL-01, a SITE AREA EMERGENCY based on a loss of all offsite power and loss of all onsite AC power to Essential Buses for 15 minutes.

B. Incorrect - With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses.

C. Incorrect - With buses 1A3 and 1A4 in a lockout condition no breakers will close onto the buses.

D. Incorrect - With Breakers 1A101 and 1A201 open no power is available to buses 1A1 and 1A2 from either the Startup or Standby Transformers.

Technical Reference(s): ARP 1C08A, A-5 & A-8
ARP 1C08B, A-3 & A-6
EAL-01
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

(1) Conditions and limitations in the facility license.

Comments:

6-10-11-NRC OK – enhanced need to change 55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	300000	2.1.7
	Importance Rating		4.7

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

Proposed Question: SRO Question # 90

The plant is operating in MODE 1 at 60% power with the following conditions:

- Main Condenser Backpressure is 4 inches Hg absolute, slowly rising
- Blue SCRAM lights are ON for Control Rods 38-31 and 42-27
- 1C05A D-6 ROD DRIFT alarms
- 1C05B F-1 SCRAM AIR HEADER HI/LO PRESSURE alarms
- 1C05A E- 3 SBLC TANK HI/LO LEVEL alarms
- Condensate Recirculation Flow Control Valve CV-1428 is SHUT

Which one of the following describes:
(1) The expected plant condition, and
(2) The appropriate mitigating procedure?

Instrument Air Header Pressure is:

A. (1) below 70 psig.
(2) It is required to enter and direct actions per AOP-518, Failure of Instrument and Service Air and IPOI-5, Reactor Scram..

B. (1) below 70 psig.
(2) It is required to enter and direct actions per AOP-518, Failure of Instrument and Service Air and AOP-644 Feedwater / Condensate Malfunction. A Reactor scram is not required at this time.

C. (1) above 80 psig.
(2) It is required to enter and direct actions per AOP-691, Condenser High Backpressure and IPOI-5, Reactor Scram.

D. (1) above 80 psig.
(2) It is required to enter and direct actions per AOP-255.1, Control Rod Movement/Indication Abnormal and AOP-644 Feedwater/Condensate Malfunction.

Proposed Answer: A

Explanation (Optional):

A. Correct - Scram Air Header Lo alarms at 68 psig, which indicates Instrument Air Header Pressure is below 80 psig. This requires AOP-518 execution. Rod Drifts require Reactor Scram.

B. Incorrect - This would be true without Rod Drift indications.

C. Incorrect - This would be true above 7.5 inches Hg without Rod Drift indications.

D. Incorrect - Rod Drift due to Instrument Air Loss is not covered by AOP-255.1. Condensate Recirculation valve failure may induce RPV Level variation.

Technical Reference(s): AOP-518 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 4071

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

6-10-11-NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	202002 A2.06	
	Importance Rating		3.3

Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Plant-Specific

Proposed Question: SRO Question # 91

The plant is operating in MODE 1 at 62% power with the following conditions:

- The plant is starting up from a maintenance outage
- Core Flow is 37 Mlbm/hr
- The "A" RFP develops a severe oil leak and trips
- RPV level lowers to 184 inches

Which one of the following is the:
(1) What are the impacts on the plant due to this event? AND
(2) What procedures must you enter and what action do you direct?

A. (1) The Recirculation Pumps will run back to 45% speed after a 15 second time delay
(2) Enter into ARP 1C06B C-3, "A" RX FEED PUMP 1P-1A TRIP OR MOTOR OVERLOAD, direct the RO to manually scram the reactor.

B. (1) The Recirculation Pumps will run back to 20% speed after a 15 second time delay
(2) Enter into ARP 1C06B C-3, "A" RX FEED PUMP 1P-1A TRIP OR MOTOR OVERLOAD, direct the RO to manually scram the reactor.

C. (1) The Recirculation Pumps will run back to 45% speed
(2) Enter AOP 644, Feedwater/Condensate Malfunction, and direct the RO to stabilize RPV level

D. (1) The Recirculation Pumps will run back to 20% speed
(2) Enter AOP 644, Feedwater/Condensate Malfunction, and direct the RO to stabilize RPV level

Proposed Answer: C

Explanation (Optional):

A. Incorrect – 45% runback has no 15 second time delay, that is on 20% runback, also, reactor scram is not required if initial reactor power is below 75% power.
B. Incorrect – 20% runback would not be activated (FW flow not <20%), also, reactor scram is not required if initial reactor power is below 75% power.
C. Correct - A 45% runback would be initiated, since only 1 RFP pump is running, with RPV level < 185”. A scram is not required since starting power is 62, AOP 644 would be entered and level stabilized.
D. Incorrect - 20% runback would not be activated (FW flow not <20%),

Technical Reference(s): ARP 1C06B, C-3
AOP 644 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	204000 2.1.30	
	Importance Rating		4.0

Conduct of Operations: Ability to locate and operate components, including local controls. (RWCU)

Proposed Question: RO Question # 92

The plant was operating in MODE 1 at 100% power when an event occurred which required a manual reactor scram. Current plant conditions are as follows:

- Reactor power is 12%
- 77 control rods NOT full in
- ATWS EOP entered
- Reactor Water Level band is 15” to 87” using FW
- Neither SBLC pump is available

Boron injection using RWCU has been directed.

(1) What procedure must be directed?
(2) What actions must be taken to inject Boron?

A. (1) Defeat 14 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C82 located at RB 812' Level

B. (1) Defeat 14 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C52 located at RB 786' Level

C. (1) Defeat 15 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C82 located at RB 812' Level

D. (1) Defeat 15 must be installed in the Control Room to override RWCU isolation signal.
(2) RWCU NRHX High Temperature override must be installed at RWCU panel 1C52 located at RB 786' Level

Proposed Answer: B

Explanation (Optional):

A. Incorrect - NRHX High Temperature is overridden at panel 1C52.

B. Correct – Defeat 14 must be installed, and NRHX High Temp is overridden at panel 1C52.

C. Incorrect - Defeat 14 must be installed, not Defeat 15. NRHX High Temperature is overridden at panel 1C52.

D. Incorrect - Defeat 14 must be installed, not Defeat 15. NRHX High Temperature is overridden at panel 1C52.

Technical Reference(s): SEP 304
Defeat 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2, 5

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201006 A2.05	
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWH) (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Out of sequence rod movement; P-Spec(Not-BWR6)

Proposed Question: SRO Question # 93

A plant startup is in progress.

- The 1C05 operators were withdrawing a group of rods from position 12 to position 24 by group notch withdrawal.
- All the rods in that group were at position 14, and the first rod in the group was being withdrawn again.

Then the solid-state timer malfunctioned, applying a continuous withdraw signal longer than the automatic protective circuitry would allow. After the Reactor Manual Control System responded as designed, the rod was identified at position 20. No alarms were received from nuclear instrumentation.

(1) Until the timer malfunction is reset, how will the ability of operators to move control rods be impacted?
(2) Must AOP 255.1, "Control Rod Movement/Indication Abnormal" be entered because the control rod qualifies as a "Mispositioned Control Rod"?

A. (1) Operators will NOT be able to select control rods due to a RMCS select block.
(2) AOP 255.1 must be entered because the rod has withdrawn far enough to qualify as a Mispositioned Rod

B. (1) Operators will NOT be able to select control rods due to a RMCS select block.
(2) AOP 255.1 need NOT be entered because the rod has NOT withdrawn far enough to qualify as a Mispositioned Rod

C. (1) Operators will be able to select control rods but a ROD OUT BLOCK (1C05B A-6) will prevent further withdrawals due to RMCS rod block.
(2) AOP 255.1 must be entered because the rod has withdrawn far enough to qualify as a Mispositioned Rod

D. (1) Operators will be able to select control rods but a ROD OUT BLOCK (1C05B A-6) will prevent further withdrawals due to RMCS rod block.
(2) AOP 255.1 need NOT be entered because the rod has NOT withdrawn far enough to qualify as a Mispositioned Rod

Proposed Answer: A

Explanation (Optional):

A. Correct - This RMCS feature protects against an unrequested continuous rod withdrawal in case the timer fails. Its protective action is to de-energize the rod Select Relays until reset. Mispositioned Rods are "two notches or greater beyond the remainder of rod group". This rod is three notches further out than the rest of the group

B. Incorrect - This RMCS feature protects against an unrequested continuous rod withdrawal in case the timer fails. Its protective action is to de-energize the rod Select Relays until reset, but the Mispositioned Rod definition excludes double notches, but not triple notches. "As described" rod movement was a triple notch

C. Incorrect - Cannot select rods and no Rod Block for this malfunction. However, the Mispositioned Rod statement is correct.

D. Incorrect - Cannot select rods and no Rod Block for this malfunction. Mispositioned Rod definition excludes double notches, but not triple notches. "As described" rod movement was a triple notch

Technical Reference(s):	SD 856.1, Rev. 4, Pages 13 &16; AOP 255.1,	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	SRO 5.02.01.06 Evaluate plant conditions and CR indications and determine if entry into AOP 255.1 is required	(As available)

Question Source:	Bank #	2005 NRC SRO #17	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2005

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.35
	Importance Rating		3.9

Conduct of Operations: Knowledge of the fuel-handling responsibilities of SRO's.

Proposed Question: SRO Question # 94

The reactor is in MODE 5 with core alterations in progress. What are:

(1) The required qualifications of the fuel handling supervisor? AND
Where must he be stationed during core alterations?

A. (1) A current DAEC SRO. He need NOT be active
(2) Must be present on the refueling bridge to directly supervise core alterations

B. (1) A current DAEC SRO. He must be active
(2) May supervise fuel movement from any location on the refuel floor

C. (1) A former SRO from a similar facility as long as an active RO is manipulating the bridge
(2) May supervise fuel movement from any location on the refuel floor

D. (1) A current DAEC SRO. He must be active
(2) Must be present on the refueling bridge to directly supervise core alterations

Proposed Answer: D

Explanation (Optional):

A. Incorrect – SRO needs to be an Active SRO

B. Incorrect – SRO needs to directly supervise by being on the Refueling Bridge

C. Incorrect – SRO needs to be an Active SRO

D. Correct – Per RFP 403, precaution 2.11, to be the Refueling SRO, you must be an active SRO, and directly supervise all core alterations.

Technical Reference(s): RFP 403 P&L 2.11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

6-10-11- NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.5
	Importance Rating		3.2

Equipment Control: Knowledge of the process for making design or operating changes to the facility.

Proposed Question: SRO Question # 95

In accordance with ACP 103.2, 10 CFR 50.59 SCREENING PROCESS, a 10CFR 50.59 screening determines if a proposed change, test or experiment requires:

A. NRC approval prior to implementation.

B. A Technical Specification Bases change after implementation.

C. A Technical Specification revision after implementation.

D. Evaluation for compliance with NRC Reg Guides.

Proposed Answer: A

Explanation (Optional):

A. Correct: 10 CFR 50.59 establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval.

B. Incorrect: A Technical Specification Bases change after implementation is not part of determination.

C. Incorrect: The evaluation may determine if a Technical Specifications is required it does not however always require a Technical Specification revision, and it would not allow a TS change after the fact for a facility change. However, an UFSAR change is sometimes allowed to take up to 2 years to make, and this choice would be plausible if the candidate transferred this rule to TS.

D. Incorrect: Does not evaluate compliance with NRC Reg Guides.

Technical Reference(s): ACP-103.2, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

6-10-11- NRC OK with change.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G3	2.3.11
	Importance Rating		3.8

Radiation Control: Ability to control radiation releases.

Proposed Question: SRO Question # 96

The plant is operating in MODE 1 at 30% power with the following conditions:

- Plant is being shutdown for a Drywell entry to find the cause of increased floor drain leakage
- Operators were about to commence an air purge (de-inerting) of the containment when both Offgas Stack Radiation Monitors, RM-4116A&B, were declared inoperable due to a failed surveillance test.
- KAMAN 9 and 10, Offgas Stack KAMAN monitors, remain in-service and operable

Which one of the following is correct regarding the operators ability to de-inert while RM-4116A&B are not operable?

De-inerting may _____.

A. NOT begin because containment venting in this situation would be an unmonitored release.

B. NOT begin because a Group 3 isolation caused by RM-4116A&B inoperability would NOT allow containment venting.

C. begin because the Offgas KAMANs being operable satisfy ODAM and Technical Specification requirements for a release.

D. begin as long as appropriate administrative controls being maintained on the containment vent and purge valves while they are open.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - Although the ARP states RM-4116A&B are required in modes 1, 2, and 3 (during venting and purging of the Primary Containment. However T.S. Section 3.3.6.1.L.2 permits the use of alternate instrumentation.

B. Incorrect - Offgas Stack Radiation Monitors, RM-4116A&B becoming inoperable do NOT cause a Group 3 isolation. Offgas Vent Pipe Radiation Monitors, RM-4116A&B Hi HI will cause a Group 3 isolation but the downscale/inoperative does NOT.

C. Incorrect - Offgas Stack Radiation Monitors, RM-4116A&B becoming inoperable do NOT cause a Group 3 isolation. Offgas Vent Pipe Radiation Monitors, RM-4116A&B Hi HI will cause a Group 3 isolation but the downscale/inoperative does NOT.

D. Correct - RM-4116A&B are required in modes 1, 2, and 3 (during venting and purging of the Primary Containment. However T.S. Section 3.3.6.1.L.2 permits establishing administrative control of the primary containment vent and purge valves using continuous monitoring of alternate instrumentation.

Technical Reference(s): 1C03A, A-4 & C-4
T.S. 3.3.6.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S. 3.3.6.1 including Table 3.3.6.1-1

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #21161

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.29
	Importance Rating		4.4

Emergency Procedures / Plan: Knowledge of the emergency plan.

Proposed Question: SRO Question # 97

The plant is operating in MODE 1 at 100% power with the following conditions:

- It is day shift
- While lowering a crate of highly radioactive material from the 5th floor, the sling breaks, causing the contents of the crate to spill out on the ground floor of the Reactor Building
- No one is injured
- Railroad Access ARM is alarming and reading 30 mR/hour

The OSM takes or directs the following actions:

- Declares a Notification of Unusual Event HU-5, based on OSM judgment.
- Sounds the Evacuation Alarm.
- Makes a Plant Page announcement for all personnel to evacuate the Reactor Building.
- Repeats the Evacuation alarm and Plant Page announcement.

Which one of the following is correct concerning the OSM's compliance with the Emergency Plan?

A. All of the OSM's actions have complied with the Emergency Plan.

B. The entire plant must be evacuated when the Evacuation Alarm is used for an EAL declaration.

C. The OSM may NOT declare an EAL based on judgment, this may only be performed by the EC when the TSC has been staffed.

D. The Evacuation Alarm is only used for EAL declarations of ALERT or greater, and may not be used for a Notification of Unusual Event.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - For accountability purposes, if we are in an emergency classification condition, evacuation of any area of the plant requires evacuation of the entire plant.

B. Correct - IAW EPIP-1.3 The OSM/CRS shall direct sounding of the Evacuation Alarm for approximately ten (10) seconds for any event classified as an ALERT or greater, it is discretionary at the NOUE classification. However for accountability purposes, if we are in an emergency classification condition, evacuation of any area of the plant requires evacuation of the entire plant.

C. Incorrect - The OSM may declare the judgment EAL

D. Incorrect - It is discretionary at the NOUE classification.

Technical Reference(s): EPIP-1.3, pg 4
EBD-H, pg 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #21162

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:
6-10-11- NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.46
	Importance Rating		4.2

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

Proposed Question: SRO Question # 98

The plant is operating in MODE 1 at 100% power with the following conditions:

- EHC Pressure Regulator "A" is in service
- The Operator At The Controls (OATC) reports a small step change (rise) of RPV pressure
- This small pressure rise is confirmed using 1C07 indications
- Reactor power is still 100% and stable
- Reactor steam dome pressure is 1021 psig and stable

Based upon these indications:

(1) The "A" pressure regulator has failed ___(1)___.
(2) The Technical Specification action required is…

A. (1) high
(2) Declare LCO 3.2.2, MCPR, NOT met, and take action to lower reactor power to less than 21.7% power within 4 hours.

B. (1) low
(2) Declare LCO 3.2.2, MCPR, NOT met, and take action to lower reactor power to less than 21.7% power within 4 hours.

C. (1) high
(2) Declare LCO 3.4.10, Reactor Steam Dome Pressure, NOT met, and take action to be in MODE 3 within 12 hours.

D. (1) low
(2) Declare LCO 3.4.10, Reactor Steam Dome Pressure, NOT met, and take action to be in MODE 3 within 12 hours.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. When one pressure regulator is out of service, OI 693.1 directs the MCPR tech spec to be entered and power to be lowered to < 21.7% within 4 hours.

B. Correct – The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. When one pressure regulator is out of service, OI 693.1 directs the MCPR tech spec to be entered and power to be lowered to < 21.7% within 4 hours.

C. Incorrect - The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. The Steam Dome Pressure TS is > 1025 psig, which is met in this question.

D. incorrect - The pressure regulator failed low, which allowed the “B” pressure regulator to take control of reactor pressure. The Steam Dome Pressure TS is > 1025 psig, which is met in this question.

Technical Reference(s): OI 693.1
TS 3.7.7 and 3.4.10
SD 693.1a (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:
6-10-11- NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.36
	Importance Rating		4.1

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO Question # 99

While supervising fuel handling activities in the Spent Fuel Pool, you discover a minor typographical error in the approved Fuel Moving Plan (FMP). Per RFP 403, PERFORMANCE OF FUEL HANDLING ACTIVITIES, which of the following describes the process for correcting the error to the fuel moving plan?

A. Minor pen & ink changes to the FMP may be made by the Fuel Handling Supervisor with concurrence from the Shift Manager and Operations Manager only.

B. Minor pen & ink changes to the FMP are not allowed. A revised plan must be submitted for review by Reactor Engineering, the Shift Manager, and the Operations Manager only.

C. Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor and the Reactor Engineer only.

D. Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager only.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - The Fuel Handling Supervisor and Reactor Engineer must also approve the change.

B. Incorrect - The questions states this is a minor change, minor changes are permitted with the proper reviews.

C. Incorrect - The Shift Manager must also approve the change.

D. Correct - Minor pen & ink changes to the FMP may be made by Reactor Engineering with concurrence from the Fuel Handling Supervisor, Reactor Engineer, and the Shift Manager.

Technical Reference(s): RFP 403, Sect. 5.1.1 e, pg 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC SRO #22624

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 6

Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments:

6-10-11-NRC OK

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.19
	Importance Rating		3.4

Equipment Control: Knowledge of maintenance work order requirements.

Proposed Question: SRO Question # 100

Which one of the following completes the note below from ACP 1408.1, Work Orders, Section 3.10, Closure:

For Safety Related Systems/Components, the Control Room Supervisor (CRS) reviewing and performing the operability testing _______________ CRS who planned the operability testing in WO instructions.

A. is allowed to be the same

B. shall be different than the

C. shall review the post maintenance testing with the

D. is allowed to change that testing by verbally informing the

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The Operations Shift Supervisor reviewing and performing the operability testing shall be different than the CRS who planned the operability testing.

B. Correct - IAW ACP-1408.1, For Safety Class SR Systems/Components, the Operations Shift Supervisor reviewing and performing the operability testing shall be different than the CRS who planned the operability testing in WO instructions.

C. Incorrect - There is no requirement for the OSS to review the test with the CRS who planned the test.

D. Incorrect - If testing requirements are to be changed, the CRS must obtain concurrence, as appropriate, from affected departments.

Technical Reference(s): ACP 1408.1, pg 52 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC #19818

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 3

Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments: