Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K1.03	
	Importance Rating	3.6	

Knowledge of the physical connections and/or cause-effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: Suppression pool

Proposed Question: RO Question # 1

Which ONE of the following correctly describes the RCIC suction transfer interlock?

The RCIC inboard and outboard Torus suction valves MO-2516 and MO-2517 open automatically on…

A. low CST level at 1' 1/4"; and, when both valves begin to open, MO-2500 CST Suction automatically closes.

B. low CST level at 1' 1/4"; and, when both valves are fully open, MO-2500 CST Suction automatically closes.

C. high Suppression Pool water level; when both valves begin to open, MO-2500 CST Suction automatically closes.

D. high Suppression Pool water level; and, when both valves are fully open, MO-2500 CST Suction automatically closes.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Torus suction valves must be full open before the MO-2500 CST Suction automatically closes. The first part of the distracter is correct, however the second part is not correct, but plausible, if the candidate assumes that both valves cycle at the same time to prevent having both valves fully open at the same time and cross connecting the CST and torus.

B. Correct - The suction path will automatically shift from the Condensate Storage Tanks to the Suppression Pool on a low level in the Condensate Storage Tanks of 1' 1/4".
The RCIC inboard and outboard torus suction valves MO-2516 and MO-2517 open automatically on low CST level, and when both valves are fully open, MO-2500 CST Suction automatically closes.

C. Incorrect – RCIC suction valves will NOT swap on a high Suppression Pool level. Plausible because HPCI suction valves will swap on a high Suppression Pool level and if the candidate assumes that both valves cycle at the same time to prevent having both valves fully open at the same time and cross connecting the CST and torus.

D. Incorrect – RCIC suction valves will NOT swap on a high Suppression Pool level, plausible because HPCI suction valves will swap on a high Suppression Pool level.

Technical Reference(s): SD 150, pgs 10, 19 & 20
OI 150, pgs 30 & 31 (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: N/A

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K1.01	
	Importance Rating	3.6	

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

Proposed Question: RO Question # 2

Following a reactor scram, the following conditions exist:

- Reactor level +190 inches
- Reactor pressure 139 psig
- Drywell pressure 1.72 psig

Based upon the given conditions, which ONE of the following Residual Heat Removal valves is closed and prevented from opening?

A. MO-2006, RHR LOOP “A” TORUS SPRAY HEADER ISOLATION

B. MO-1908, RHR SHUTDOWN COOLING ISOLATION VALVE

C. MO-2007, RHR LOOP “A” TORUS COOLING AND TEST RETURN HDR ISOLATION

D. MO-1940, RHR HX 1E-201B BYPASS VALVE

Proposed Answer: B

Explanation (Optional):

A. Incorrect – For the given conditions MO-2006 is able to be opened, the valve is isolated when containment pressure is > 2 psig. Plausible because the candidate may assume that a high Drywell pressure is needed to place Torus sprays in service.

B. Correct - Of the signals listed, only the reactor pressure signal causes an RHR isolation/interlock. This high-pressure interlock prevents the SDC section of piping from being over pressurized. A reactor pressure of approximately 135 psig (per ARP 1C03B B-4 this pressure is approximately 100 psig) initiates an isolation of SDC suction valves MO-1908 and 1909. The LPCI piping is also protected from over pressurization, but the setpoint is 450 psig.

C. Incorrect – For the given conditions MO-2007 is able to be opened, the valve is isolated when containment pressure is > 2 psig. Plausible because the candidate may assume that a high Drywell pressure is needed to place Torus sprays in service.

D. Incorrect – MO-1940 has an automatic open function on a LPCI initiation signal. It does NOT have an auto close function.

Technical Reference(s): ARP 1C05B, D-8
SD-149 pgs. 31-34 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K2.01	
	Importance Rating	2.8	

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

Proposed Question: RO Question # 3

Backup Instrument Air Compressor 1K1 is in STANDBY operating mode with power being supplied from the preferred source.

An electrical disturbance occurs resulting in:

- LLRPSF transformers XR1 and XR2 are de-energized
- A Bus 1A3 lockout

Which ONE of the following is the response of the Air Compressor 1K1?

Air Compressor 1K1 will…

A. start when header pressure reaches 90 psig and will cycle to maintain 90 - 100 psig.

B. start when header pressure reaches 100 psig and will cycle to maintain 100 - 110 psig.

C. NOT start until its power supply is manually transferred from the normal source to the alternate source.

D. NOT start until HSS-3002, BACKUP COMPRESSOR 1K-1 PRESSURE SELECT SWITCH is placed to the PRIMARY position.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - The electrical power supply to the backup compressor is lost due to the 1A3 lockout. Air Compressor 1K1 power source will NOT start and must be shifted to its' alternate power source.

B. Incorrect - The electrical power supply to the backup compressor is lost due to the 1A3 lockout. Air Compressor 1K1 power source will NOT start and must be shifted to its' alternate power source.

C. Correct - The electrical power supply to the backup compressor is from essential bus, 1B33 or 1B45. Bus 1B33, the normal source, is powered from Essential Bus 1A3, a lockout will prevent the 1G31 Diesel from supplying the bus therefore Air Compressor 1K1 power source must be shifted to its' alternate power source. The power source is selected from either 1B33 or 1B45, the alternate source, via a manually selectable power transfer switch 1N3312.

D. Incorrect - Pressure select switch HSS-3002 is a two position switch used to determine the operating mode of the backup compressor. The backup compressor is normally maintained in the STANDBY status. It will NOT have an effect until the power supply is transferred to the alternate source.

Technical Reference(s): OI-518.1, Precaution and Limitation 19 and Note in Sect 4.7, SD-518 pg. 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank (19111)22681

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 DAEC (#4)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K2.01	
	Importance Rating	3.1	

Knowledge of electrical power supplies to the following: Major D.C. loads

Proposed Question: RO Question # 4

The plant is at rated conditions with the electric plant in its normal full power lineup when a fault results in a loss of 125 VDC bus 1D20.

While in this condition, a loss of coolant accident occurs and the reactor scrams.

Following the scram, the Main Turbine is manually tripped.

Which one of the following is correct regarding high pressure systems or components that are available to control RPV level from the control room?

A. HPCI and both Reactor Feed Pumps

B. RCIC and both Reactor Feed Pumps

C. HPCI and “A” Reactor Feed Pump ONLY

D. RCIC and “A” Reactor Feed Pump ONLY

Proposed Answer: D

Explanation (Optional):

A. Incorrect: HPCI will not function without 125 VDC Div II power (bus 1D20). Plausible if the candidate confuses the power supplies of HPCI and RCIC (Div 1 versus Div 2). Additionally only the “A” RFP will be available because bus 1A2 will de-energize when the turbine is tripped due to the loss of control power. Plausible if the candidate does not understand the response of the electric plant to a loss of DC following a turbine trip.

B. Incorrect: Only the “A” RFP will be available because bus 1A2 will de-energize when the turbine is tripped due to the loss of control power.

C. Incorrect: HPCI will not function without 125 VDC Div II power (bus 1D20). Plausible if the candidate confuses the power supplies of HPCI and RCIC (Div 1 versus Div 2).

D. Correct: RCIC will still be available with a loss of 125 VDC. Although RCIC suction will transfer to the torus and one channel of isolation logic is disabled, it is still available for level control. Additionally when the turbine trips, bus 1A2 will de-energize due to the loss of control power supplied by Div II 125 VDC. However, 1A1 still has control power and will transfer to the off-site power source which will supply power to the “A” RFP.

Technical Reference(s): AOP 302.1, page 10 (Attach if not previously provided)

ARP 1C04C, B-9 for effects on a loss of “B” logic (Div 2)

ARP 1C03C, C-9 for effects on a loss of “B” logic (Div 2)

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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K3 01	
	Importance Rating	3.8	

Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: Reactor water level

Proposed Question: RO Question # 5

The plant is operating at 100% power when a small “A” recirc loop leak results in drywell pressure rising to 3 psig. The following conditions exist:

- The reactor is scrammed and all rods are fully inserted
- RPV level is stabilized at 190 inches range using high pressure injection systems
- Drywell pressure remains at 3 psig
- RPV pressure is 900 psig and slowly lowering
- The CRS directs that both Core Spray pumps be manually shutdown
- The BOP operator:
  - Momentarily places both Core Spray pump control switches to the STOP position and then releases the switch.
  - The Core Spray pump amber OVERRIDE INITIATED lights above both pump control switches illuminate.

Then, the “A” recirc loop ruptures, resulting in the following:

- RPV level and pressure lower rapidly
- Drywell pressure rises and stabilizes at ~ 22 psig.

Which one of the following is correct regarding the ability of Core Spray to assist in water level recovery?

The Core Spray pumps will _______________ when RPV pressure lowers to less than 450 psig.

A. NOT automatically restart, but can be manually re-started and then will inject

B. automatically re-start when RPV level lowers to less than +64 inches and inject

C. automatically re-start IF the CORE SPRAY INITIATION SEALED-IN reset pushbuttons are first depressed and will inject regardless of RPV level

D. NOT automatically restart, but can be manually restarted IF the CORE SPRAY INITIATION SEALED-IN reset pushbuttons are first depressed and then will inject

Proposed Answer: A

Explanation (Optional):

A. Correct: Placing the core spray pump control switch to STOP with an auto start signal present seals out the auto start signal and lights an amber light located above the pump control switch indicating that the auto start signal has been overridden. The core spray pump may be restarted using the START position of the control switch.

B. Incorrect: The Core Spray pumps will not automatically restart. Plausible if the candidate believes that the redundant pump auto start signal of RPV level < 64" in conjunction with RPV pressure < 450 psig will restart the pump.

C. Incorrect: The Core Spray pumps will not automatically restart. Plausible if the candidate believes that the pumps will restart if the initiation reset pushbuttons are depressed. With drywell pressure above 2 psig, the initiation logic will not reset if the reset pushbuttons are depressed, therefore the core spray pumps will not restart. Although, the injection valves will open when RPV pressure lowers below 450 psig.

D. Incorrect: The core spray pump may be restarted using the START position of the control switch. Depressing the seal-in pushbutton is not required. Plausible in that this pushbutton would reset the initiation and the pump override, but ONLY IF the initiation signals are clear. With the high drywell pressure condition the initiation cannot be reset.

Technical Reference(s): SD 151, pg 9
OI 518.1, P & L 19 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 K3.02	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: Reactor manual control: Plant-Specific

Proposed Question: RO Question # 6

During a reactor startup the following conditions exist:

- The reactor is critical
- IRM "C" is on range 2
- All other IRMs are on range 3
- All SRM detectors are partially withdrawn
- SRM readings are all approaching 10^5 cps

Which ONE of the following is the effect of stuck withdraw button causing SRM "A" to withdraw from the core and its count rate to lower to 90 cps?

A. ONLY the retract permitted light on SRM/IRM drive controls will go out.

B. ONLY the detector retracted when NOT permitted annunciator will alarm.

C. ONLY the detector retracted when NOT permitted annunciator will alarm, and a rod block will prevent further rod motion.

D. ONLY a detector retracted when NOT Permitted annunciator will alarm, a rod block will prevent further rod motion, and a half scram will occur.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - withdrawing the SRM out past 90 will cause a rod withdrawal block if the SRM detectors are NOT fully inserted and SRM flux level is lower than 100 cps when the IRMs are on range 1 or 2.

B. Incorrect - withdrawing the SRM out past 90 will cause a rod withdrawal block if the SRM detectors are NOT fully inserted and SRM flux level is lower than 100 cps when the IRMs are on range 1 or 2.

C. Correct - A rod withdrawal block will occur if the SRM detectors are NOT fully inserted and SRM flux level is lower than 100 cps when the IRMs are on range 1 or 2.

D. Incorrect - withdrawing the SRM out past 90 will cause a rod withdrawal block if the SRM detectors are NOT fully inserted and SRM flux level is lower than 100 cps when the IRMs are on range 1 or 2. There are NO half scrams on the SRMs for normal operating conditions and even if the shorting links were installed the low reading on the SRM would NOT cause a scram.

Technical Reference(s): ARP 1C05A, E-5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K4.06	
	Importance Rating	3.6	

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Redundant power sources to vital buses

Proposed Question: RO Question # 7

The plant is operating at 100% power. Essential Buses 1A3 and 1A4 are being supplied from the Standby Transformer.

The Startup transformer has just been re-energized and is available.

Which ONE of the following describes the actions that will occur if the Standby Transformer Lockout Relay energizes to trip the "Standby Transformer Feeder Breaker, CB8490 (M Breaker)"?

Essential Buses1A3 and 1A4 …

A. de-energize and transfer to the Startup Transformer.

B. remain energized due to fast transfer to the Startup Transformer.

C. de-energize and are repowered by the Standby Diesel Generators.

D. are de-energized with the Standby Diesel Generators running but NOT connected.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – There NO is transfer to the Startup Transformer during this event. Plausible because the Startup Transformer also supplies the Essential Buses. Candidate may assume that the Startup Transformer will pick up the load.

B. Incorrect - The conditions are not met for a fast transfer. The DGs will re-energize the load centers. Plausible because the Startup Transformer also supplies the Essential Buses. Candidate may assume that the Startup Transformer will pick up the load.

C. Correct – The “Standby Transformer Feeder Breaker CB8490” breaker is the supply to the standby transformer. The loss of the Standby Transformer results in a loss of power to both vital buses. The EDGs will pick up the load centers on the bus undervoltage start signal.

D. Incorrect - Standby Transformer Lockout Relay will “Lock Out” the Standby Transformer the Essential Load Centers 1A3 and 1A4 are not locked out and will be re-energized by the SBDGs. Plausible because the Essential Buses Supply Breakers 1A301 and 1A401 trip open deenergizing Buses 1A3 and 1A4 the candidate may assume the lockout extends to the Essential Buses or that the SBDGs will NOT automatically close in on the dead buses.

Technical Reference(s): SD 304 pgs. 24-29
ARP 1C08B (A11) (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K4.02	
	Importance Rating	2.6	

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Charcoal bed decay heat removal

Proposed Question: RO Question # 8

Both SBGT Trains have been in operation following a LOCA in the Drywell. After some time the lineup was changed as follows:

- "B" SBGT Train remains in service
- "A" SBGT Train has been placed in STANDBY
- "A" SBGT Train, Carbon Bed Temperature is 220°F and rising slowly

At this time, which ONE of the following actions is required by OI 170, Standby Gas Treatment System, for the "A" SBGT Train?

A. Dispatch the fire brigade to manually start the fire deluge sprays.

B. Initiate a manual cooldown using the Cooldown /Outside Air Valve.

C. Dispatch the fire brigade to verify the automatic initiation of the fire deluge sprays.

D. Place the “A” SBGT Train back in service and verify Carbon Bed Temperatures lower.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The operator has the option of manually starting the fire deluge sprays when Carbon Bed Temperatures reach 255°F. The temperature given in the stem 220°F is well below this.

B. Incorrect – The deluge system automatically initiates when charcoal temperatures reach 310°F.

C. Correct - Per OI 170, Section 9.1. Manual Cooldown of SBGT System Train, Open COOLDOWN/OUTSIDE AIR VLV AV-5801A and Place HS-5825A INTAKE VALVE to CLOSE. Then monitor the CARBON BED TEMP Indicator TI-5838A on Panel 1C24A occasionally over the next 30 minutes to verify the carbon bed temperature remains within reasonable limits (between 150°F and 255°F).

D. Incorrect – The “A” SBGT Train should NOT be returned to service the train must be shutdown, cooled and monitored over the next 30 minutes to verify the carbon bed temperature remains within reasonable limits (between 150°F and 255°F).

Technical Reference(s): OI 170 Section 9.1, pgs. 22-23 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

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

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

Proposed Question: RO Question # 9

The plant is at power and operators are performing post maintenance testing on the HPCI turbine in order to exit the HPCI LCO.

HPCI has been manually started in the CST-CST mode using the manual control and adjusting the pulsar knob on the front of the flow controller. The system is stable at 3500 rpm and 2000 gpm.

At this point, the operator throttles open CV-2315, TEST BYPASS, to achieve 3000 GPM.

Which ONE of the following correctly describes the response of HPCI Discharge Pressure and turbine speed?

PI-2306 (HPCI) PUMP DISCHARGE PRESSURE indication on 1C03 _____(1)_____.
SI-2284, HPCI TURBINE SPEED indication on 1C03 ______(2)______.

A. (1) Rises
(2) Rises

B. (1) Lowers
(2) Lowers

C. (1) Rises
(2) Remains the same

D. (1) Lowers
(2) Remains the same

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Pump laws dictate that a pump at constant speed will have a lower discharge pressure at higher flow. HPCI speed does not change because the control signal remains the same until the manual control knob is adjusted.

B. Incorrect - HPCI speed does not change because the control signal remains the same until the manual control knob is adjusted.

C. Incorrect - Discharge pressure will not rise but will lower with a constant speed and higher flow.

D. Correct – Pump laws dictate that a pump at constant speed will have a lower discharge pressure at higher flow. HPCI speed does not change because the control signal remains the same until the manual control knob is adjusted.

Technical Reference(s): SD 152, page 11 & 12 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

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

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

Proposed Question: RO Question # 10

The plant was operating at 100% when the following events occurred:

- A transient occurred which resulted in a total loss of feedwater
- A large unisolable RWCU leak OUTSIDE the Drywell has occurred
- HPCI is unavailable due to planned maintenance

Assuming no operator action, which ONE of the following correctly states ALL the necessary conditions for the ADS Valves to open?

The ADS logic will open the ADS Valves when any low pressure ECCS Pump…

A. breaker is closed and RPV water level reaches Low-Low-Low.

B. reaches normal discharge pressure and RPV water level reaches Low-Low-Low.

C. breaker is closed, and RPV water level reaches Low-Low-Low, and two minutes have elapsed.

D. reaches normal discharge pressure, and RPV water level reaches Low-Low-Low, and two minutes have elapsed.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – This would be true if Pump breakers were used by logic and two minute timer didn't delay actuation

B. Incorrect – This would be true if two minute timer didn't delay actuation

C. Incorrect – This would be true if Pump breakers were used by logic

D. Correct - ECCS Pump Discharge Pressure signal is used for ADS Logic, when RPV water level reaches Low-Low-Low and two minutes elapse, ADS Valves will OPEN.

Technical Reference(s): OI-183.1, pg 7
SD-183-1 pg 9 & 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 11320

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Duane Arnold

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K6.02	
	Importance Rating	3.6	

Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: 24/48 volt D.C. power: Plant-Specific

Proposed Question: RO Question # 11

The plant is starting up with the following:

- Mode Switch is in STARTUP
- IRMs are on Range 6

24 VDC power to IRM "E" is then lost.

Which ONE of the following describes the effect of the power loss?

A. NO effect.

B. ONLY a rod block will occur.

C. ONLY a half scram will occur.

D. A rod block and half scram will occur.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - With a loss of 24 VDC, the indication fails downscale and the IRM becomes inoperative, causing a rod block and half scram.

B. Incorrect - With a loss of 24 VDC, the IRM becomes inoperative, causing a rod block and half scram.

C. Incorrect - With a loss of 24 VDC the indication fails downscale.

D. Correct - With a loss of 24 VDC, the indication fails downscale and the IRM becomes inoperative, causing a rod block and half scram.

Technical Reference(s): AOP-375, pgs 3 & 6
OI-878.2, pg 3
SD-878.2, pg 17
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	6
	55.43	

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

Comments:

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

Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: D.C. power: Plant-Specific

Proposed Question: RO Question # 12

The plant is operating at 100% 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 (Optional):

A. Incorrect – The "A" logic does NOT have control power, there is no alternate power to the “A” logic.

B. Correct – Because either logic system is capable of producing ADS initiations in each of the four ADS valves the loss of DC power to logic "A” has no affect on ADS operation.

C. Incorrect – All the ADS SRVs have an alternate source of control power. The loss of DC power to logic "A” has no affect on ADS operation and all four ADS SRVs open.

D. Incorrect – The "A" logic does NOT have control power, there is no alternate power to the “A” logic. The loss of DC power to logic "A” has no affect on ADS operation and all four ADS SRVs open.

Technical Reference(s): SD 183.1, pg 22
1C03A, C-6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank 20458
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 A1.11	
	Importance Rating	3.4	

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: System status lights and alarms

Proposed Question: RO Question # 13

Following a reactor scram the Scram Discharge Volume High Water Level Bypass Switch was placed in BYPASS to reset the scram and was left in that position.

After the scram was reset, the following alarms cleared:

- 1C05B (D-1) SCRAM DISCHARGE VOLUME NOT DRAINED
- 1C05B (C-1) SCRAM DISCHARGE VOLUME HI LEVEL TRIP

Then the mode switch is placed in Startup.

Which ONE of the following describes the position of the Scram Discharge Volume Vent and Drain Valves (1) after resetting the scram and (2) after the mode switch is placed in Startup?

A. (1) Open
(2) Open

B. (1) Open
(2) Closed

C. (1) Closed
(2) Open

D. (1) Closed
(2) Closed

Proposed Answer: A

Explanation (Optional):

A. Correct. The scram discharge volume high water level bypass switch only bypasses the SDV high level scram it does nothing regarding the position of the SDV vent and drain valves. Therefore when the scram is reset the SDV vent and drain valves open and will remain open even after the scram bypass switch is taken out of bypass.

B. Incorrect - Once the mode switch is in startup the scram bypass is removed and the valves remain open.

C. Incorrect - the SDV vent and drain valves will be open before and open after the mode switch is placed in Startup

D. Incorrect - the SDV vent and drain valves will be open before the mode switch is placed in Startup.

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

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank 48127

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A1.07	
	Importance Rating	3.0	

Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: APRM (gain adjustment factor)

Proposed Question: RO Question # 14

The plant is operating at 100% power. The Process Computer indicates the following information:

	1	2	3	4	5	6
APRM READING	101.5	99.4	99.2	101.4	101.0	99.8
APRM GAF	0.967	1.002	1.009	0.987	0.986	1.001

Which ONE of the following provides the correct response?

The indication from APRMs ___(1)___ are conservative because thermal power is ___(2)___ indicated power.

A. (1) 1, 4, 5
(2) less than

B. (1) 2, 3, 6
(2) greater than

C. (1) 2, 3, 6
(2) less than

D. (1) 1, 4, 5
(2) greater than

Proposed Answer: A

Explanation (Optional):

A. Correct - The APRMs at all times should read no lower than actual core power. If the AGAF is greater than 1.0 then the APRM needs adjustment. Thus any AGAF >1.0 would be non-conservative with respect to power operations because actual thermal power would be greater than indicated power.

B. Incorrect – APRMs 2, 3, 6 are NOT conservative and for the GAF to be conservative the actual power must be less than the indicated power.

C. Incorrect - Incorrect – APRMs 2, 3, 6 are NOT conservative.

D. Incorrect - For the GAF to be conservative the actual power must be less than the indicated power.

Technical Reference(s): SD-878.3, pgs 38 & 39 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	WTSI 10732	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2005 Monticello

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

10 CFR Part 55 Content:	55.41	6
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 A2.03	
	Importance Rating	3.2	

Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. power failures

Proposed Question: RO Question # 15

An ATWS has occurred resulting in the need to inject boron using Standby Liquid Control (SBLC).

- A loss of Instrument AC 1Y11 occurs
- The operator places the STANDBY LIQUID CONTROL Switch HS-2613 in the PUMPS A and B RUN position

(1) How will the SBLC system and indications respond?
(2) What actions are required?

A. (1) Both pumps indicate they are running and squib valve continuity lights extinguish. SBLC tank level, pump discharge pressure and flow will indicate zero.
(2) Inject boron into the RPV with RWCU (SEP 304).

B. (1) Both pumps indicate they are running and squib valve continuity lights extinguish. SBLC tank level, pump discharge pressure and flow will indicate zero.
(2) Send an operator to the SBLC system to monitor parameters.

C. (1) Both pumps indicate they are shutdown and the squib valve continuity lights are illuminated. Pump discharge pressure and flow will indicate zero. SBLC tank level will indicate normally.
(2) Inject boron into the RPV with RWCU (SEP 304).

D. (1) Both pumps indicate they are shutdown and the squib valve continuity lights are illuminated. Pump discharge pressure and flow will indicate zero. SBLC tank level will indicate normally.
(2) Send an operator to the SBLC system to monitor parameters

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Both pumps and squib valves operate normally (powered from 1B34 and 1B44) there is no need for alternate boron injection

B. Correct - Per AOP 317, The loss of Instrument AC will result in a loss of power to: SBLC Storage Tank Level (LI-2600A), SBLC Pump Discharge Pressure (PI-2605), SBLC System Flow (FI-2620), Injection Valve Position (V26-0032). Therefore, flow and tank level indication will fail to zero. The pumps & squib valves have NOT lost power (powered from 1B34 and 1B44). There is NO need for alternate boron injection and IAW the ARP for the low SBLC tank level the control room should send an operator to the SBLC system to monitor the system parameters.

C. Incorrect – Both pumps and squib valves operate normally (powered from 1B34 and 1B44), the loss of 1Y11 will cause SBLC tank level, flow and pressure to indicate zero. There is NO need for alternate boron injection and IAW the ARP for the low SBLC tank level the control room should send an operator to the SBLC system to monitor the system parameters.

D. Incorrect - Both pumps and squib valves operate normally (powered from 1B34 and 1B44), the loss of 1Y11 will cause flow and pressure to indicate zero

Technical Reference(s): AOP 317, pg 10
ARP 1C05A (E-3) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 46971(1410)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A2.03	
	Importance Rating	2.9	

Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low CCW temperature

Proposed Question: RO Question # 16

DAEC has just completed refueling and is back on line. The following additional conditions exist:

- The Fuel Pool Cooling (FPC) system is in its normal configuration IAW OI 435, Fuel Pool Cooling System
- The FPC Heat Exchanger (HX) outlet temperature is slowly rising due to the recently discharged fuel

Based on the above, which ONE of the following is correct regarding:

(1) the impact of the FPC HX outlet temperature reaching 130°F

AND

(2) the action taken in order to mitigate the rising temperature?

A. (1) Temperature limit on the Filter/Demin resin will be exceeded.
(2) Throttle close the FPC HX bypass valve as required to increase flow through the heat exchanger.

B. (1) Temperature limit on the Filter/Demin resin will be exceeded.
(2) Throttle open the FPC HX RBCCW outlet isolation valve as required to increase cooling to the heat exchanger.

C. (1) Temperature limit on the Spent Fuel Pool will be exceeded.
(2) Throttle close the FPC HX bypass valve as required to increase flow through the heat exchanger.

D. (1) Temperature limit on the Spent Fuel Pool will be exceeded.
(2) Throttle open the FPC HX RBCCW outlet isolation valve as required to increase cooling to the heat exchanger.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The Heat Exchanger bypass valve is a normally closed valve. Temperature is controlled by throttling the RBCCW flow to the heat exchanger. Plausible if the candidate believes that the HX bypass valve is normally in mid position to control system temperature.

B. Correct: The temperature limit to preclude resin damage is 130 degrees. Temperature control is via throttling the RBCCW cooling flow through the heat exchanger.

C. Incorrect: The temperature limit on the Spent Fuel Pool will not be exceeded till 150 degrees. Additionally, the temperature is controlled by throttling the RBCCW flow to the heat exchanger.

D. Incorrect: The temperature limit on the Spent Fuel Pool will not be exceeded till 150 degrees.

Technical Reference(s): SD 435, page 9 (Attach if not previously provided)

OI 435, section 4.0, Normal Operation of the FPC System, page 12.

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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	203000	A3.02	
	Importance Rating	4.0		

Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Pump start

Proposed Question: RO Question # 17

The plant is operating at 100% power when a Reactor SCRAM on High DRYWELL PRESSURE occurs simultaneously with a STARTUP and STANDBY TRANSFORMER Lockout. Both Standby Diesel Generators START as designed and come up to rated speed and voltage.

- Reactor Water Level: 100 inches and rising slowly
- Reactor Pressure: 800 psig and lowering slowly

Which of the following describe the ECCS response to these conditions?

After the SBDG Output Breakers CLOSE, ...

A. "A" and "B" RHR Pumps START after 5 seconds
"C" and "D" RHR Pumps START after 10 seconds
"A" and "B" Core Spray Pumps START after 15 seconds

B. "A" and "C" RHR Pumps START after 5 seconds
"B" and "D" RHR Pumps START after 10 seconds
"A" and "B" Core Spray Pumps START after 15 seconds

C. "A" and "B" Core Spray Pumps START after 5 seconds
"A" and "B" RHR Pumps START after 10 seconds
"C" and "D" RHR Pumps START after 15 seconds

D. "A" and "B" Core Spray Pumps START after 5 seconds
"A" and "C" RHR Pumps START after 10 seconds
"B" and "D" RHR Pumps START after 15 seconds

Proposed Answer: C

Explanation (Optional):

A. Incorrect - The CS pumps in both subsystems are automatically started approximately 5 seconds after AC power is available. The RHR C and D pumps approximately 15 seconds after AC power is available. The RHR A and B pumps approximately 10 seconds after AC power is available

B. Incorrect - The CS pumps in both subsystems are automatically started approximately 5 seconds after AC power is available. The RHR C and D pumps approximately 15 seconds after AC power is available. The RHR A and B pumps approximately 10 seconds after AC power is available

C. Correct - Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started approximately 5 seconds after AC power is available.
Upon receipt of an initiation signal, all four LPCI pumps are automatically started (A and B pumps approximately 10 seconds after AC power is available, and C and D pumps approximately 15 seconds after AC power is available).

D. Incorrect - The CS pumps in both subsystems are automatically started approximately 5 seconds after AC power is available. The RHR C and D pumps approximately 15 seconds after AC power is available. The RHR A and B pumps approximately 10 seconds after AC power is available

Technical Reference(s): SD 149, pg 20
SD 151, pg 18
OI 149, pg 11
OI 151, pg 6
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank 46127
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A3.03	
	Importance Rating	2.5	

Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: SPDS/ERIS/CRIDS/GDS: Plant-Specific

Proposed Question: RO Question # 18

A plant startup and heatup is in progress at 600 psig RPV pressure. The Main Turbine is in Chest Warming. IDT computer terminals are set up as follows:

- The terminal next to 1C14 is being used to monitor KAMAN parameters.
- The Chemist is collecting data from the MIDAS terminal.
- The terminal to the left of the ANSOE desk is being used as the alarm monitor as usual.

For the remaining terminals, which set of Plant Process Computer displays meets the requirement for this plant condition?

A. TURBINE (Key turbine parameters)
Group Display 15 (APRM and other data)
STATUS (Plant status overview display)

B. HEATUP (9 Reactor Temperature parameters)
Group Display 32 (Startup parameters)
POWER (Turbine parameter display)

C. SPDS Containment Conditions
DR (Rod position display)
Group Display 3 (Turbine vibration data)

D. SCR (Strip Chart Request output points)
PPC Menu (Top Level Menu)
Group Display 33 (Turbine Startup parameters)

Proposed Answer: C

Explanation (Optional):

A. Incorrect - All options are groups of normal displays used in the control room. Plausible because they all contain displays that may be used at times during reactor startup / turbine warm-up, but incorrect because they do not contain SPDS displays.

B. Incorrect - All options are groups of normal displays used in the control room. Plausible because they all contain displays that may be used at times during reactor startup / turbine warm-up, but incorrect because they do not contain SPDS displays.

C. Correct - "SPDS should be continuously displayed in the Control Room. Any of the SPDS displays (top level, 2nd level, 3rd level or graphs) are acceptable. Containment Conditions is an SPDS display that displays containment isolation valve status.

D. Incorrect - All options are groups of normal displays used in the control room. Plausible because they all contain displays that may be used at times during reactor startup / turbine warm-up, but incorrect because they do not contain SPDS displays.

Technical Reference(s): OI-831.4, P&L #5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank

Modified Bank # 48747 (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 A4.01	
	Importance Rating	2.8	

Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source

Proposed Question: RO Question # 19

The plant is at 100% power when the following sequence occurs:

- UPS Inverter 1D45 output voltage lowers to 80% of nominal output voltage
- Ten seconds later, UPS Inverter 1D45 output voltage recovers to its nominal output voltage

Which ONE of the following is correct regarding:

(1) The response of the UPS

AND

(2) Any associated control room indications?

UPS distribution panel 1Y23 is currently being supplied by…

A. (1) Regulating Transformer 1Y4
(2) Both feedwater regulating valves are locked up due to the "break before make" transfer

B. (1) Regulating Transformer 1Y4
(2) There are no adverse indications in the control room due to the "make before break" transfer

C. (1) Static Inverter 1D45
(2) Both feedwater regulating valves are locked up due to the "break before make" transfer

D. (1) Static Inverter 1D45
(2) There are no adverse indications in the control room due to the "make before break" transfer

Proposed Answer: D

Explanation (Optional):

A. Incorrect: When Inverter output voltage lowered, the Static Transfer switch transferred the power supply to the Regulating Transformer 1Y4. When Inverter output voltage recovered, it transferred back to the Inverter. These transfers are “make before break” and would have no adverse impact on plant systems. Plausible if the candidate believes that a manual transfer back to the inverter is required. Additionally if the candidate believes it is a “break before make” transfer, both feed reg valves would lock up (see AOP 357, Loss of 120 VAC UPS)

B. Incorrect: When Inverter output voltage lowered, the Static Transfer switch transferred the power supply to the Regulating Transformer 1Y4. When Inverter output voltage recovered, it transferred back to the Inverter. Plausible if the candidate believes that a manual transfer back to the inverter is required.

C. Incorrect: The transfers are “make before break” and would have no adverse impact on plant systems. Plausible if the candidate believes the transfer is similar to that of transfer switch 1Y22 which is a “break before make” transfer.

D. Correct: When Inverter output voltage lowered, the Static Transfer switch transferred the power supply to Regulating Transformer 1Y4. When Inverter output voltage recovered, it transferred back to the Inverter. The transfers are “make before break” resulting in continuous power to 1Y23 distribution panel.

Technical Reference(s): SD 357, pages 10 and 11 (Attach if not previously provided)

AOP 357, page 2

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: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A4.02	
	Importance Rating	3.4	

Ability to manually operate and/or monitor in the control room: Synchroscope

Proposed Question: RO Question # 20

OI 324, Standby Diesel Generators, Section 6.5, PARALLELING THE "A" SBDG SYSTEM TO ESSENTIAL BUS 1A3, is being performed.

Which ONE of the following describes what the operator must ensure during this evolution?

Adjust the INCOMING VOLTS SYNCHRONIZE to slightly ____(1)____ than RUNNING VOLTS SYNCHRONIZE.

Using the A DIESEL GENERATOR 1G-31 SPEED ADJUST, adjust diesel generator speed to a slow, ____(2)____ synchroscope rotation.

A. (1) lower
(2) clockwise

B. (1) higher
(2) clockwise

C. (1) lower
(2) counter clockwise

D. (1) higher
(2) counter clockwise

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Incoming voltages must be raised to slightly more than running volts.

B. Correct - Incoming voltages must be raised to slightly more than running volts and the diesel speed adjusted to where the Synchroscope is rotating slowly in the clockwise direction.

C. Incorrect - Incoming voltages must be raised to slightly more than running volts and the diesel speed adjusted to where the Synchroscope is rotating slowly in the clockwise direction.

D. Incorrect - the diesel speed adjusted to where the Synchroscope is rotating slowly in the clockwise direction.

Technical Reference(s): OI-324, Sect 6.5, pg 33 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 1738

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Susquehanna

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 2.4.50	
	Importance Rating	4.2	

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

Proposed Question: RO Question # 21

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

- 1C05A (D-1), REACTOR VESSEL HI/LO LEVEL RECORDER ALARM actuates.
- "B" GEMAC RPV Water Level Indicator LI-4560 indicates 198 inches, RISING.
- "A" and "C" GEMAC RPV Water Level Indicators LI-4559 and 4561 indicates 186 inches, LOWERING.
- Reactor Vessel Level Control is in “3” ELEMENT with “B” Level selected.
- Total Steam Flow is 7.0 Mlbm/hr.
- Total Feedwater Flow is 6.5 Mlbm/hr.

Which ONE of the following is correct regarding the ACTUAL RPV Water Level trend and the action that will stabilize RPV Water Level?

ACTUAL RPV Water Level is:

A. RISING; place the 1-ELEMENT 3-ELEMENT Control Switch in the 1-ELEMENT position.

B. LOWERING; place the 1-ELEMENT 3-ELEMENT Control Switch in the 1-ELEMENT position.

C. RISING; select A-LEVEL on REACTOR WATER LEVEL CONTROL INPUT SELECT HSS-4560.

D. LOWERING; select A-LEVEL on REACTOR WATER LEVEL CONTROL INPUT SELECT HSS-4560.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - Level is lowering, this response is plausible and would be a correct trend and action for a Feedwater Flow input failed LOW.

B. Incorrect – Level is lowering, this response with level rising is plausible and would be a correct trend and action for a Steam Flow input failed HIGH.

C. Incorrect - This is plausible; would be true for mismatched Steam Flow and Feed Flow with no disparity between level channels.

D. Correct - With LI-4559 rising and redundant channels LI-4560 and LI-4561 lowering, level channel failure is indicated. Additionally with steam flow greater than feedwater flow the lowering level is substantiated. This will be corrected by swapping level channels as directed in the ARP.

Technical Reference(s): ARP 1C05A D-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 4078

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Duane Arnold

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 2.4.3	
	Importance Rating	3.7	

Emergency Procedures / Plan: Ability to identify post-accident instrumentation. (Uninterruptable Power Supply)

Proposed Question: RO Question # 22

Which ONE of the following is a post-accident instrument that will lose power during a loss of 120 VAC Instrument Control Power Panel 1Y11?

A. NMR-9253, SRM COUNTS

B. LI-4565C, FUEL ZONE RX LEVEL

C. PI-1816A, CRD CHARGING WATER PRESS

D. PI-4563, A PRESSURE (REACTOR PRESSURE)

Proposed Answer: B

Explanation (Optional):

A. Incorrect - SRM COUNTS NMR-9253 is NOT a post accident instrument and this instrument is powered from 120 VAC UNINTERRUPTIBLE POWER

B. Correct - During a loss of instrument AC Panel 1Y11 FUEL ZONE RX LEVEL LI-4565C fails downscale activating 2/3 core coverage interlock.

C. Incorrect - Although PI-1816A, CRD CHARGING WATER PRESS is powered from Instrument AC it is not a post accident instrument

D. Incorrect - (REACTOR PRESSURE) A PRESSURE PI-4563 is powered from 120 VAC UNINTERRUPTIBLE POWER

Technical Reference(s): AOP-317, pg 4 & 11
T.S. Table 3.3.3.1-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:			(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:		Last NRC Exam:	N/A

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

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 2.1.7	
	Importance Rating	4.4	

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

Proposed Question: RO Question # 23

During the watch turnover it is noted that the Standby Transformer is energized and Standby Transformer Breaker (M), OCB-8490, is closed. Later, during a panel walkdown, it is noted that the Standby Transformer Breaker (M), OCB-8490, RED and GREEN indicating lights are NOT lit. BOTH light bulbs are GOOD.

All other 1C08 panel indications are normal for this plant configuration, and no annunciators were received from this event.

Which ONE of the following describes the impact of this indication?

A. Essential bus power will not transfer to the Standby Transformer when required.

B. OCB-8490 has lost control power and will ONLY respond to breaker protective breaker trips.

C. OCB-8490 has lost control power and will NOT OPEN when the control switch is taken to the TRIP position

D. Essential Bus Power will ONLY slow transfer to the Standby Transformer when required.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - This is plausible however, this condition would not inhibit the ability of the essential busses to transfer to the Standby Transformer. The failure is in the Standby Transformer Feeder Breaker not the essential bus logic.

B. Incorrect - This is plausible because 125 VDC supplies control power to the breaker. However a loss of power to the tripping circuit would block all remote and protective trips.

C. Correct – The breaker indicating lights are powered from breaker control power, lost continuity in the control circuit de-energizes indicating lights. A loss of power to the tripping circuit would block all remote and protective trips.

D. Incorrect – This is plausible however, this condition would not have any effect on the ability of the essential busses to transfer to the Standby Transformer. The failure is in the Standby Transformer Feeder Breaker not the essential bus logic.

Technical Reference(s): OI-304.2, Sect 2 (2), pg 4
SD-304 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 11277
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

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

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

Proposed Question: RO Question # 24

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

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

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

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

Diesel Generator Room Ventilation Supply Fan 1V-SF-20 (1V-SF-21) starts and ventilation dampers align to control room temperature and pressure _____(3)_____ after the SBDG picks up the 1A3 and 1A4 buses..

A. (1) ≤10 seconds
(2) immediately
(3) immediately

B. (1) ≤10 seconds
(2) immediately
(3) 5 seconds

C. (1) ≤18.5 seconds
(2) 5 seconds
(3) immediately

D. (1) ≤18.5 seconds
(2) 5 seconds
(3) 5 seconds

Proposed Answer: A

Explanation (Optional):

A. Correct - The time frame assumed in the UFSAR and the Technical Specifications for the SBDGs to start and reenergize the essential busses is ≤10 seconds). The ESW pump and SBDG room ventilation start immediately when power is restored to the bus. The 18.5 seconds used as a distracter is based on the 8 to 8.5 second delay to start the SBDG during a degraded voltage situation. In this question the stem specifies a complete instantaneous loss of offsite power, therefore the time delay does NOT apply.

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

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

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

Technical Reference(s): SD 324, pg 36, 38
OI 324, pg 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 46985

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

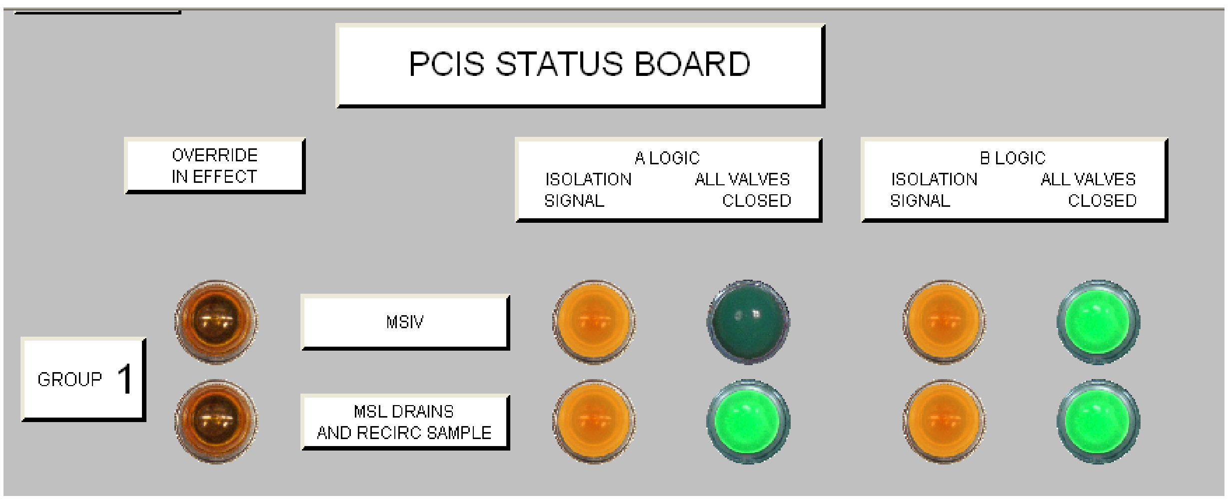
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A4.04	
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: System indicating lights and alarms. (Primary Containment Isolation System /Nuclear Steam Supply Shut-Off)

Proposed Question: RO Question # 25

The plant is operating at 12% power. A plant transient occurred; and several seconds later, the CIMS panel indicates the following:



Which ONE of the following correctly describes a condition which would cause the above indications?

A. Drywell pressure rising above 2.5 psig, and there has been a failure of CV4412 "A" MAIN STEAM LINE INBOARD ISOLATION to isolate.

B. A reactor high water level tripped the main turbine, and there has been a failure of CV4419 "C" MAIN STEAM LINE OUTBOARD ISOLATION to isolate.

C. Steam tunnel temperatures rising above 200°F for several seconds, and there has been a failure of CV4420 "D" MAIN STEAM LINE INBOARD ISOLATION to isolate.

D. Turbine building main steam line area temperatures rising above 200°F for several seconds, and there has been a failure of CV4416 "B" MAIN STEAM LINE OUTBOARD ISOLATION to isolate.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - DW press > 2# will NOT provide a Group 1 isolation signal; Plausible: failure of an INBOARD MSIV to close will provide an incomplete "A" Logic indication for the MSIVs. Also, 2.5 psig in DW would result in Group 2 & 3 isolation signals.

B. Incorrect - There is NOT a Group 1 isolation signal from a reactor high water level. Failure of an OUTBOARD MSIV to close will NOT provide an incomplete "A" Logic indication for the MSIVs.

C. Correct - Steam Tunnel temps > 200°F provide a Group 1 isolation signal; Failure of an INBOARD MSIV to close will provide an incomplete "A" Logic indication for the MSIVs.

D. Incorrect - Plausible – Turbine building main steam line area temps > 200°F provide a Group 1 isolation signal; however, failure of an OUTBOARD MSIV to close will NOT provide an incomplete "A" Logic indication for the MSIVs.

Technical Reference(s):	50007_57-05_lp, pages 24-27 and 40-42 SD 959.1, All conditions verified using DAEC Simulator 4/10/12 Print this question in COLOR	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	DAEC Bank # 48708	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K1.03	
	Importance Rating	2.5	

Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant air systems: Plant-Specific

Proposed Question: RO Question # 26

The plant is operating at 100% power.

A loss of Instrument Air occurred to the Standby Liquid Control (SBLC) tank bubbler dip tube.

Which ONE of the following correctly describes the effect on indicated SBLC tank level and SBLC heater operation?

The control room Indicated SLC tank level will fail ___ (1) ___.
The SBLC Tank Heater will ____ (2) ____.

A. (1) low
(2) trip

B. (1) high
(2) trip

C. (1) low
(2) NOT trip

D. (1) high
(2) NOT trip

Proposed Answer: C

Explanation (Optional):

A. Incorrect - A low level in the SLC Tank will NOT automatically trip the heaters.

B. Incorrect – SLC Tank level indication will fail low. A low level in the SLC Tank will NOT automatically trip the heaters.

C. Correct - Instrument Air provides compressed air for storage tank level bubbler instrumentation via PCV- 2600 and FIC-2600. In the event of a Loss of Instrument Air, air-flow to the SBLC Storage Tank bubbler (level indication) will be lost. As a result, indicated storage tank level will decrease (or fail low). A low level in the SLC Tank will NOT automatically trip the heaters.

D. Incorrect - Incorrect – SLC Tank level indication will fail low.

Technical Reference(s): OI 153, pg 7
AOP 518, pg 8
SD 153, pgs 8 & 27
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 48408

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290002 K1.01	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between REACTOR VESSEL INTERNALS and the following: Main steam system

Proposed Question: RO Question # 27

Which ONE of the following prevents the Reactor Pressure Vessel Steam Dryer from lifting as steam flow rises?

A. Hold-down ties which attach the steam separator and dryer to the core shroud.

B. Its weight, and analysis that high dryer ΔPs are limited by the length of the dryer skirt.

C. Hold-down brackets fitted to the underside of the vessel head contact the top of the dryer.

D. Its weight, and that steam flow is directed perpendicular both up and down within the dryer.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Only the steam separator is attached to the core shroud.

B. Incorrect - The dryer is restricted from lifting by hold-down brackets fitted to the underside of the vessel head.

C. Correct - The dryer rests on support brackets attached to the reactor vessel wall and is restricted from lifting by hold-down brackets fitted to the underside of the vessel head.

D. Incorrect – Regardless of the steam flows the ΔP would try to lift the dryer.

Technical Reference(s): SD-262, pgs 27-28
LP-50007-262, pg 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:			(As available)
Question Source:	Bank #	DAEC Bank # 22356	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	Not Used
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	2	
	55.43		

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 K2.03	
	Importance Rating	2.7	

Knowledge of electrical power supplies to the following: Recirculation system valves

Proposed Question: RO Question # 28

The plant is at 100% power with the electric plant aligned for normal full power operation.

Then the following occurs:

- A load reject scram and complete loss of offsite power occurs.
- "A" Diesel Generator fails to start.
- "B" Diesel Generator starts and re-energizes bus 1A4.

What is the status of the power supply to the Recirc Loop Suction, Discharge and Discharge Bypass valves?

		Suction Valve	**Discharge Valve**	**Discharge Bypass Valve**
A.	Loop "A"	De-energized	Energized	Energized
	Loop "B"	De-energized	Energized	Energized
B.	Loop "A"	De-Energized	De-Energized	De-Energized
	Loop "B"	Energized	Energized	Energized
C.	Loop "A"	Energized	Energized	Energized
	Loop "B"	Energized	Energized	Energized
D.	Loop "A"	De-Energized	De-Energized	De-Energized
	Loop "B"	De-energized	Energized	Energized

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The Recirc pump suction valves will also have power. Plausible in that “LPCI swing bus” 1B34A/1B44A supplies power to all LPCI and RHR valves that may have to reposition during a LPCI Loop selection. However the Recirc loop suction valves which do not reposition are also powered from 1B34A/1B44A. This would be correct if this answer if this were not the case.

B. Incorrect: “LPCI swing bus” 1B34A/1B44A supplies power to all Recirc Loop valves. The bus is normally aligned to 1A3 via 1B34. When 1A3 failed to re-energize following the loss of off-site power, a power seeking ABT circuit transfers the “LPCI swing bus” to 1A4 via 1B44. Plausible if the candidate is believes that the Recirc and LPCI loop select valves are divisional in configuration.

C. Correct: “LPCI swing bus” 1B34A/1B44A supplies power to all Recirc Loop valves. The bus is normally aligned to 1A3 via 1B34. When 1A3 failed to re-energize following the loss of off-site power, a power seeking ABT circuit transfers the “LPCI swing bus” to 1A4 via 1B44.

D. Incorrect: Incorrect: “LPCI swing bus” 1B34A/1B44A supplies power to all Recirc Loop valves. The bus is normally aligned to 1A3 via 1B34. When 1A3 failed to re-energize following the loss of off-site power, a power seeking ABT circuit transfers the “LPCI swing bus” to 1A4 via 1B44. Plausible if the candidate believes that the Recirc and LPCI loop select valves are divisional in configuration and that the recirc suction valves are not powered from this bus.

Technical Reference(s): SD 304, page 34 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	219000 K3.01	
	Importance Rating	3.9	

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE will have on following: Suppression pool temperature control

Proposed Question: RO Question # 29

The plant is operating at 90% power.

- HPCI is running to support a pump operability surveillance
- Torus Cooling is in service IAW OI 149, RHR System

Which ONE of the following describes:

(1) How Torus water temperature is controlled, IAW OI 149, during this evolution?

AND

(2) How Torus water temperature would be affected if an inadvertent LPCI initiation signal occurred during the HPCI surveillance test?

A. (1) Throttle MO-2031 [1941] HEAT EXCH OUTLET
(2) Torus water temperature would rise

B. (1) Close, Open or Throttle MO-2030 [1940] A[B] HEAT EXCH BYPASS
(2) Torus water temperature would rise

C. (1) Throttle MO-2031 [1941] HEAT EXCH OUTLET
(2) Torus water temperature would remain the same or lower

D. (1) Close, Open or Throttle MO-2030 [1940] A[B] HEAT EXCH BYPASS
(2) Torus water temperature would remain the same or lower

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The RHR heat exchanger outlet valves (MO-2031 [1941]) are isolation valves and NOT used for throttling flow, torus cooling is controlled by closing or opening MO-2030 [1940] A[B] HEAT EXCH BYPASS. Plausible because the RHR Heat Exchanger INLET valve is used for controlling the cooldown rate in shutdown cooling mode as stated in OI 149, "Control the cooldown rate by adjusting MO-2030 [MO-1940] A[B] HEAT EXCH BYPASS valve and MO-2029 [MO-1939] A[B] HX INLET THROTTLE valve on 1C03".

B. Correct – (1) Per OI 149, Section 5.4, Open or Close MO 2030 [1940] A[B] HEAT EXCH BYPASS valve as required.
(2) Per SD 149 - The LPCI initiation signal overrides all modes of the RHR System (except shutdown cooling). The intent is to direct maximum system effort toward restoring and maintaining the reactor vessel water level, i.e., all pumps are started, all non LPCI modes secured. This would secure torus cooling and the resulting HPCI surveillance test would cause torus temperature to rise.

C. Incorrect - The RHR heat exchanger outlet valves (MO-2031 [1941]) are isolation valves and NOT used for throttling flow. Plausible because the RHR Heat Exchanger INLET valve is used for controlling the cooldown rate in shutdown cooling mode as stated in OI 149, "Control the cooldown rate by adjusting MO-2030 [MO-1940] A[B] HEAT EXCH BYPASS valve and MO-2029 [MO-1939] A[B] HX INLET THROTTLE valve on 1C03. Torus cooling is controlled by closing or opening MO-2030 [1940] A[B] HEAT EXCH BYPASS". A LPCI signal would shut the Torus Cooling valves and Torus temperature would rise.

D. Incorrect - A LPCI signal would shut the Torus Cooling valves and Torus temperature would rise.

Technical Reference(s): OI 149, Sect 5.4
SD 149, pg 21 (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: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 K4.07	
	Importance Rating	2.5	

Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Timing of rod insert and withdrawal cycles (rod movement sequence timer)

Proposed Question: RO Question # 30

During a control rod single notch withdrawal, a Reactor Manual Control System timer malfunction occurs resulting in the withdraw bus remaining energized.

Which ONE of the following RMCS protective features prevents a continuous withdrawal event AND how is this accomplished?

A. When a withdrawal signal lasts greater than 2 seconds, the current rod is de-selected and a select block is generated.

B. When the Rod Movement control switch is held in "withdraw" for greater than 2 seconds, a rod out block is generated.

C. When the Rod Movement control switch is momentarily placed in "withdraw," any stopping of the rod sequence timer generates a select block.

D. When a single notch withdrawal signal is generated, the rod position indication system initiates a rod block if the control rod moves beyond one notch.

Proposed Answer: A

Explanation (Optional):

A. Correct - The operation of the RMCS solid state timer is monitored in the rod withdrawal mode to protect against unrequested continuous rod withdrawal should the timer fail, holding in the withdraw time interval. If the drive out time interval during notch withdrawal is actuated for a period of approximately 2 seconds (normal interval is 1.5 seconds) the timer malfunction circuit de-energizes the rod select relays (enforces a select block). Actuation of the timer malfunction circuit energizes the amber SELECT BLOCK indicator light.

B. Incorrect – The Rod Movement control switch may be held in "withdraw" indefinably only the time the timer is energized is monitored.

C. Incorrect – Rod withdrawal accidents are prevented by monitoring the amount of time the rod sequence timer energizes the withdraw bus.

D. Incorrect – Rod withdrawal accidents are prevented by monitoring the amount of time the rod sequence timer energizes the withdraw bus, the RPIS does not cause this block or prevent this event.

Technical Reference(s): SD 856.1, 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 Bank # 19307

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002 K5.01	
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM: Fluid coupling: BWR-3,4

Proposed Question: RO Question # 31

Regarding the operation of a Recirc MG Set scoop tube:

With the Scoop Tube fully withdrawn,

___(1)___ (more / less) oil is present in the fluid coupler, resulting in

___ (2) ___ (raising / lowering) Recirc pump speed.

A. (1) less
(2) raising

B. (1) more
(2) lowering

C. (1) less
(2) lowering

D. (1) more
(2) raising

Proposed Answer: C

Explanation (Optional):

A. Incorrect: When there is less oil in the coupler, recirc speed lowers.

B. Incorrect: Withdrawing the scoop tube reduces the amount of oil.

C. Correct: When the scoop tube moves away from the shaft, the amount of oil being removed increases. This decreases the amount of oil in the working circuit. The lower amount of oil decreases the hydraulic coupling and decreases the generator speed.

D. Incorrect: Withdrawing the scoop tube reduces the amount of oil. This in turn will cause speed to lower.

Technical Reference(s): SD 264, pages 24 & 25, Figure 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	DAEC Bank # 48329	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content:	55.41	6
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 K6.03	
	Importance Rating	2.8	

Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: Temperature Compensation

KA Justification: Pressure Compensation compensates for changes in the density of the coolant in the RPV across the spectrum of operating pressures. Since the vessel is saturated temperature and pressure compensation are equivalent.

Proposed Question: RO Question # 32

The RPV pressure instruments used to pressure compensate the Nuclear Boiler Instrumentation are not available.

Which range of RPV Level indicators will be affected?

A. Fuel Zone

B. Wide Range Yarway

C. Narrow Range Yarway

D. Narrow Range GEMAC

Proposed Answer: A

Explanation (Optional):

A. Correct: A modification was installed to pressure-compensate the fuel zone detectors to correct the inaccuracies caused by the difference in variable leg density from the calibrated cold condition to normal operating and accident pressures. Two amber lights on 1C-03, located next to the fuel zone level indicators, will illuminate if sensed RPV pressure is not within 0-1500#.

B. Incorrect: The Wide Range Yarways are not pressure compensated by sensing RPV pressure. Plausible in that they do have heated reference legs to reduce the error imparted by changes in coolant density between shutdown and operating conditions.

C. Incorrect: The Narrow Range Yarways are not pressure compensated by sensing RPV pressure. Plausible in that the Wide Range Yarways (but not the Narrow Range) do have heated reference legs to reduce the error imparted by changes in coolant density between shutdown and operating conditions.

D. Incorrect: The GEMACs are not pressure compensated.

Technical Reference(s): SD 880, page 17 (Attach if not previously provided)

P&ID M115

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: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 A1.02	
	Importance Rating	2.8	

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: CRD drive pressure

Proposed Question: RO Question # 33

The plant is operating at rated conditions.

Which ONE of the following identifies the component manipulations that will raise CRD drive water header differential pressure indication at Panel 1C05?

A. Adjust CRD SYSTEM FLOW CONTROL FC-1814 to open the flow control valve or throttle open DRIVE WATER ΔP CONTROL MO-1830

B. Adjust CRD SYSTEM FLOW CONTROL FC-1814 to open the flow control valve or throttle closed DRIVE WATER ΔP CONTROL MO-1830

C. Adjust CRD SYSTEM FLOW CONTROL FC-1814 to close the flow control valve or throttle open DRIVE WATER ΔP CONTROL MO-1830

D. Adjust CRD SYSTEM FLOW CONTROL FC-1814 to close the flow control valve or throttle closed DRIVE WATER ΔP CONTROL MO-1830

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Throttling open MO-1830 will lower CRD drive pressure differential.

B. Correct - To raise drive water pressure close MO-1830 or open FC-1814.

C. Incorrect. Throttling closed FC-1814 will lower CRD drive pressure differential.

D. Incorrect. Throttling closed FC-1814 will lower CRD drive pressure differential.

Technical Reference(s): OI 255, pg 9
SD 255, pg 6 & 32 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	WTSI 4396	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2008 Brunswick

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

10 CFR Part 55 Content:	55.41	5
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 A2.11	
	Importance Rating	2.8	

Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Offgas system low flow

Proposed Question: RO Question # 34

With the plant operating at 100% power the following conditions occur:

- SJAE flow on FR-1374 has lowered
- Off Gas flow rates on FR-4132 have become erratic
- Annunciator 1C34, C-1, RECOMBINER HI/LO TEMP is in alarm
- Reactor power has been lowered to 25%

Which ONE of the following is (1) the cause of these conditions and besides monitoring critical parameters (2) what other action is required?

A. (1) An Offgas premature recombination event has occurred.
(2) Shutdown the hydrogen water chemistry system.

B. (1) An Offgas premature recombination event has occurred.
(2) Start the mechanical vacuum pump and shutdown the steam jet air ejectors.

C. (1) Steam supply pressure to the Offgas Jet Compressor has lowered to 290 psig.
(2) Start the mechanical vacuum pump to stabilize condenser vacuum.

D. (1) Steam supply pressure to the Offgas Jet Compressor has lowered to 290 psig.
(2) Place the second set of SJAE in service in parallel with the operating SJAE.

Proposed Answer: A

Explanation (Optional):

A. Correct - In the event of Offgas premature recombination, off gas flow and recombiner temperature would lower. Action must be taken to stop the premature recombination to prevent damage to other off gas equipment and piping. To limit H2 the hydrogen water chemistry system must be secured.

B. Incorrect – This is plausible, because with a Offgas premature recombination with power <10% the procedure directs start the mechanical vacuum pump to stabilize condenser vacuum, however in this case power level is 25%.

C. Incorrect – The low SJAE flow and erratic offgas flow indicate a H2 event NOT an offgas isolation. This distracter is plausible, because with a Offgas isolation with power <10% the procedure directs starting the mechanical vacuum pump to stabilize condenser vacuum, however in this case power level is 25%.

D. Incorrect – The low SJAE flow and erratic offgas flow indicate a H2 event NOT an offgas isolation. This distracter is plausible, because with the low off-gas flow another set of air ejectors would provide additional flow.

Technical Reference(s): AOP-672.3 (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: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000 A3.02	
	Importance Rating	3.2	

Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including: Alarm and indicating lights

Proposed Question: RO Question # 35

A control rod is being withdrawn from position 24 to position 26.

Which of the following conditions will result in a Control Rod Drift alarm? Consider all conditions separately.

Condition 1: The reed switch for position 25 fails to close.

Condition 2: The reed switch for position 25 fails to open.

Condition 3: The reed switch for position 26 fails to close.

A. Condition 2 ONLY

B. Condition 3 ONLY

C. Conditions 1 and 2 ONLY

D. Conditions 2 and 3 ONLY

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Condition 3 will also cause a rod drift alarm. Once the timer times out a rod drift alarm will occur for that rod if an even numbered reed switch is not closed.

B. Incorrect: Condition 2 will also cause a rod drift alarm. A rod drift alarm will occur if after the timer times out an odd numbered reed switch is closed, as this may be indicative of a rod drifting.

C. Incorrect: Condition 1 will not cause a rod drift alarm. Odd numbered reed switches are normally open.

D. Correct: If an even numbered (latched position) reed switch is not closed and the rod is not being driven, a drift alarm will be generated. A drift alarm will be also be generated if an odd numbered (between latched positions) reed switch closes without the rod being driven. As soon as the timer times out, i.e., not being selected and driving, a drift alarm will occur from either condition 2 or 3.

Technical Reference(s): SD 856.1, page 28 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000 A4.13	
	Importance Rating	2.9	

Ability to manually operate and/or monitor in the control room: Turbine inlet pressure

Proposed Question: RO Question # 36

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

A steam leak occurs on the sensing line for the "A" Pressure Regulator such that "Steam Throttle Pressure A" is slowly failing DOWNSCALE.

Assuming NO operator action is taken, which ONE 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 (Optional):

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. No scram will occur.

B. Incorrect - 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. No scram will occur.

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 rise will be seen by the “B” regulator. So 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 at a slightly elevated reactor and throttle pressure.

Technical Reference(s): AOP 262, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 19235

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000 2.4.31	
	Importance Rating	4.2	

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: RO Question # 37

Following a failure to scram, the RO is performing RIP 103.2, Increase CRD Cooling Flow and Pressure. The RO has placed both CRD pumps in service when the following annunciators alarm and remain in alarm:

- 1C06A (A-12) “A” CONDENSATE PUMP 1P-8A TRIP OR MOTOR OVERLOAD
- 1C06A (A-13) “B” CONDENSATE PUMP 1P-8A TRIP OR MOTOR OVERLOAD

Which ONE of the following describes the effects of these annunciators on RIP 103.2?

A. The CRD pumps will trip on a loss of suction pressure; the RIP must be exited.

B. Continue in the RIP securing one CRD pump to prevent both pumps tripping on a loss of suction.

C. Continue in the RIP; the CRD pump suction source will automatically shift to the condensate storage tanks.

D. Exit the RIP, manually shift the CRD pump suction to the condensate storage tanks, and re-enter the RIP.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The CRD pumps will NOT trip on a loss of suction. The CST suction is always lined up to the CRD pump suction. If CV-1497 closes the CRD pump suction will automatically shift to the CSTs. Plausible because the normal suction source is isolated on a trip of both condensate pumps.

B. Incorrect – There is no reason to secure a CRD pump. Plausible because the normal suction source is isolated on a trip of both condensate pumps and the normal source has a higher pressure than the alternate source from the CSTs.

C. Correct - The CRD pumps will NOT trip on a loss of suction. These annunciators indicate the condensate pump has tripped or is about to trip. The condensate pump breaker closed position provides an interlock to CV-1497 through which condensate flows from the reject line to the Control Rod Drive Hydraulic System. However the CST suction is always lined up to the CRD pump suction. If CV-1497 closes the CRD pump suction will automatically shift to the CSTs.

D. Incorrect – There is no reason to exit the RIP both CRD pumps will remain running with the CSTs as their suction. Plausible because the normal suction source is isolated on a trip of both condensate pumps.

Technical Reference(s): 1C06A, C-12 & C-13
RIP 103.2
SD 639, pg 9
SD 255, pg 6
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 K5.10	
	Importance Rating	3.2	

Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Withdraw error: P-Spec(Not-BWR6)

Proposed Question: RO Question # 38

Given the following:

- A reactor startup is in progress with reactor power in the source range
- The RWM is in OPERATE enforcing the Rod Sequence
- There are NO Rod Worth Minimizer (RWM) errors currently existing
- The control rod sequence directs that control rod 26-15 be selected and withdrawn from position 10 to position 12
- Control rod 26-15 is the 2nd rod in rod step 27
- Rod step 27 contains 8 control rods
- Step 27 control rods
    - Insert Limit: 8
    - Withdraw Limit: 12

When control rod 26-15 is withdrawn, the rod "double-notches" and settles at position 14.

Which ONE of the following is correct regarding further control rod movement?

The RWM will automatically block…

A. ANY control rods from being withdrawn. ALL control rods can be inserted until three insert errors are created.

B. ANY control rods from being inserted or withdrawn. Control rod 26-15 can ONLY be repositioned after bypassing the RWM.

C. control rod 26-15 from further withdrawal, but it can be inserted back to position 12 to clear the rod block. NO other control rod movement is possible unless the RWM is bypassed.

D. control rod 26-15 from further withdrawal. The remaining rods in step 27 can be withdrawn within the limits of the step. Control rod 26-15 must be inserted before leaving Step 27.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: All rod movement is inhibited unless the withdraw error is corrected. Plausible in that normally, operation can continue if an insert error is made provided that there are no more than three insert errors.

B. Incorrect: The error rod can be inserted to correct the withdrawal error.

C. Correct: A withdrawal error was generated when rod 26-15 was withdrawn past it's withdraw limit. Until the error is corrected, all other control rod motion is inhibited.

D. Incorrect: A withdrawal error was generated when rod 26-15 was withdrawn past it's withdraw limit. Until the error is corrected, all other rod motion is inhibited. Plausible in that the RWM does not enforce how the rods are withdrawn or inserted within the step provided the insert and withdrawal limits are not violated.

Technical Reference(s): SD 878.8, pg 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI 12965

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK1.02	
	Importance Rating	4.1	

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Reactor vessel integrity

Proposed Question: RO Question # 39

Which ONE of the following is the highest steam dome pressure that will ensure that the reactor coolant system will remain intact at its most limiting location?

A. 1250 psig

B. 1335 psig

C. 1380 psig

D. 1590 psig

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Plausible in that this is the design pressure of the RPV.

B. Correct: The Tech Spec Safety Limit on reactor steam dome pressure protects the reactor coolant system against over-pressurization. The most limiting transient location is the recirc suction piping which has a design pressure of 1150 psig. USAS, Nuclear Power Piping Code, Section B31.1, 1967 Edition (Ref. 6) permits a maximum pressure transient of 120% of design pressure or 1380 psig (120% x 1150 psig = 1380 psig). This value is adjusted to account for the added pressure due to elevation sensed at the lowest points in the reactor pressure vessel and reactor coolant system piping in order to determine the most limiting steam dome pressure. This results in a safety limit of 1335 psig steam dome pressure.

C. Incorrect: Plausible in that 1380 psig is the maximum allowed transient pressure allowed for in the recirc loop suction piping. However it does not take into account the weight of the water above the lowest section of the reactor coolant system piping in order to determine the required steam dome pressure.

D. Incorrect: Plausible in that 1590 is the maximum allowed transient pressure rating of the recirc discharge piping. However it is not the most limiting pressure within the reactor coolant system.

Technical Reference(s):	Safety Limit 2.1.2 and associated bases.	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	WTS Bank # 295025	Adjusted for DAEC
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2010 NMP2

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

10 CFR Part 55 Content:	55.41	3
	55.43	

Mechanical components and design features of reactor primary system.

Comments:

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

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

Proposed Question: RO Question # 40

Following a loss of coolant accident the following conditions exist:

- “A” RHR Pump is in Torus cooling at 3300 gpm
- “B” RHR Pump is in Torus cooling at 4700 gpm
- Torus pressure is 5 psig
- Torus water temperature is 215 °F

NPSH limits are currently being exceeded for…

A. Both RHR Pumps

B. Neither RHR Pump

C. “A” RHR pump ONLY

D. “B” RHR pump ONLY

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The limit for “A” RHR pump is ~ 217 degrees. Plausible if the candidate sums the 2 flows and uses the “2 pumps” values for the bottom axis of the curve. If so, the limit for 8000 gpm at 5 psig torus pressure would be 215 degrees. This would be incorrect in that the two pumps are in different loops.

B. Incorrect: The “B” pump is exceeding its limit. At 4700 gpm and with a torus pressure of 5 psig, the limit is ~ 214 degrees. Plausible if the candidate again uses the “2 pump” values on the horizontal axis but considers the flow rates individually. In this case the limits for both pumps would be around 220 degrees.

C. Incorrect: The “A” RHR pump limit is ~ 217 degrees. Plausible if the candidate misinterprets the graph for the two flow rates.

D. Correct: The “B” pump is exceeding its limit. At 4700 gpm and with a torus pressure of 5 psig, the limit is ~ 214 degrees. The “A” RHR pump is < its limit of ~ 217 degrees.

Technical Reference(s): EOP Graph 8 (Attach if not previously provided)
Bases EOP Curves and Limits
Pgs. 60-61

Proposed References to be provided to applicants during examination: EOP Graph # 8, RHR NPSH

Learning Objective: LP 95.01, LO 95.00.00.17 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AK1.02	
	Importance Rating	3.3	

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

Proposed Question: RO Question # 41

The plant is in MODE 4 with “A” loop of RHR in Shutdown Cooling (SDC).

- Both Recirc pumps are secured and unavailable
- The “B” Loop of RHR is tagged out for maintenance

Then…

(1) A leak on the discharge of “A” Loop of RHR occurs
(2) RPV level lowers and SDC isolates

IAW with AOP 149, Loss of Decay Heat Removal, which ONE of the following is required to ensure adequate mixing of the coolant in the core?

Raise and control reactor water level between…

A. 170 and 211 inches.

B. 186 and 195 inches.

C. 230 and 240 inches.

D. 258 and 270 inches.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: AOP-149 requires level to be raised and controlled between 230 and 240 inches. Plausible in that an EOP 1 entry condition has occurred and this is the initial control band.

B. Incorrect: AOP-149 requires level to be raised and controlled between 230 and 240 inches. Plausible in that this range is between the low and high level alarm setpoints.

C. Correct: Insufficient RHR flow while in SDC can result in RPV thermal stratification. To prevent thermal stratification a natural circulation flow path must be established that allows the hotter water in the reactor core to flow upwards out the steam separators into the annulus region. AOP 149, Follow-up Action # 4 requires level to be raised and controlled between 230 and 240 inches. Insufficient RHR flow while in SDC can result in RPV stratification. AOP 264, pg 8 states If RWCU System is not available OR Recirc Pumps are not available, raise RPV level to greater than (>) 214" (preferably between 230" and 240") to improve natural circulation. T.S. Bases 3.4.8 RHR Shutdown Cooling System also states “Alternate methods of reactor coolant circulation that can be used include (but are not limited to) raising reactor water level above the minimum natural circulation level (i.e., lowest turnaround point for water in the steam separator).”

D. Incorrect: AOP-149 requires level to be raised and controlled between 230 and 240 inches. Plausible in that the higher the level the more the natural circulation. However this would flood the main steam lines.

Technical Reference(s): AOP 149, Follow-up Action # 4
RHR lesson plan pg 29
AOP 264, Pg 8
T.S. Bases 3.4.8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: STG AOP 149, 94.01.01.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK2.13	
	Importance Rating	4.1	

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: ARI/RPT/ATWS: Plant-Specific

Proposed Question: RO Question # 42

Which ONE of the following is the effect of tripping the Alternate Rod Insertion (ARI) system reactor water level A and C detectors?

A. One ARI system valve will open, and both recirculation pumps trip immediately.

B. Both ARI system valves will open, and one recirculation pump trips immediately.

C. One ARI system valve will open, and both recirculation pumps trip after a 9 second time delay.

D. Both ARI system valves will open, and one recirculation pump trips after a 9 second time delay.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The recirculation pumps trip after a 9 second time delay.

B. Incorrect: Only one ARI valve will open and the recirculation pumps trip after a 9 second time delay.

C. Correct: The two ATWS-RPT/ARI logics are arranged such that a trip of either logic trips both recirculation pumps (one RPT breaker per pump) and energizes one ARI solenoid valve.

D. Incorrect: Only one ARI valve will open. Additionally both recirc pumps will trip.

Technical Reference(s): SD-358, pg 23 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:	LP 358 - Reactor Protection, 22.00.00.05	(As available)

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

Question History:	Last NRC Exam:	N/A

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

10 CFR Part 55 Content:	55.41	6
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AK2.08	
	Importance Rating	2.8	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Plant ventilation

Proposed Question: RO Question # 43

Which ONE of the following describes an effect of a loss of instrument air to the Standby Gas Treatment (SBGT) system?

As a result of the loss of instrument air, SBGT...

A. will automatically start and establish rated flow.

B. will NOT automatically start; however, it can be manually started.

C. will NOT automatically start and can NOT be started manually.

D. will automatically start; however, the suction and discharge dampers will NOT open.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: SBGT will NOT auto start. SBGT will only auto start on the following signals:
Reactor Building ventilation exhaust high radiation of 8 mR/hr (inc) or downscale.
Refueling Pool exhaust high radiation of 8 mR/hr (inc), or INOP.
Primary containment (Drywell) high pressure of ≥2 psig.
Reactor vessel low water level of ≤170 inches.
Offgas Vent Pipe (Stack) HI-HI radiation.
Plausible in that the candidate may feel that SBGT will automatically start to maintain RX building D/P when RX building ventilation dampers fail closed as a result of the loss of air.

B. Correct: SBGT will NOT automatically start as a result of the loss of air. SBGT can be manually started and is normally manually started on a loss of IA (see AOP 518).

C. Incorrect: SBGT can be manually started and is normally manually started on a loss of IA (see AOP 518). Plausible in that the candidate may feel that SBGT will automatically start to maintain RX building D/P when RX building ventilation dampers fail closed as a result of the loss of air.

D. Incorrect: SBGT will NOT automatically start as a result of the loss of air. Plausible in that a loss of IOA to the RBHVAC system will cause a loss of reactor building D/P. However SBGT is designed to operate in this condition to maintain building D/P. Flow will be established because the suction and discharge dampers fail open and the cool down air damper fails closed.

Technical Reference(s): SD 170, SBGT, page 18
AOP 518, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 730, LO 67.01.01.10, e (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK2.01	
	Importance Rating	3.1	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Battery charger

Proposed Question: RO Question # 44

DAEC was operating at 100% power when an event occurred, resulting in the following:

- Breaker 1D10 ckt 4, supply to 125VDC Distribution Panel 1D13 tripped.
- 10 seconds later, a complete loss of Offsite power occurred.

Which of the following correctly describes the SOURCE of power to the remaining station 125 VDC loads one minute later?

A. A battery is supplying Division 1 DC loads
A battery is supplying Division 2 DC loads

B. A battery is supplying Division 1 DC loads
A charger is supplying Division 2 DC loads

C. A charger is supplying Division 1 DC loads
A battery is supplying Division 2 DC loads

D. A charger is supplying Division 1 DC loads
A charger is supplying Division 2 DC loads

Proposed Answer: B

Explanation (Optional):

A. Incorrect: 1A4 will be re-energized by 1G21 “B” SBDG, and therefore 1D22, Div 2 125VDC Charger will supply the Div 2 125VDC loads.

B. Correct: The complete loss of offsite power results in loss of both essential busses and start signals to both SBDGs. The loss of 1D13 results in loss of control power to the 1A3 Bus supply breakers including 1G31 “A” SBDG Output Breaker. This results in 1A3 failing to re-energize, therefore 1B32, which supplies 1D12, Div 1 125VDC Charger, is also de-energized, resulting in Div 1 loads being supplied by the battery. 1A4 will be re-energized by 1G21 “B” SBDG, and therefore 1D22, Div 2 125VDC Charger will supply the Div 2 125VDC loads.

C. Incorrect: Div 1 125VDC Charger is de-energized, resulting in Div 1 loads being supplied by the battery. 1A4 will be re-energized by 1G21 “B” SBDG, and therefore 1D22, Div 2 125VDC Charger will supply the Div 2 125VDC loads.

D. Incorrect: Div 1 125VDC Charger is de-energized, resulting in Div 1 loads being supplied by the battery.

Technical Reference(s): AOP 302.1, rev 52, p 12 (Attach if not previously provided)

AOP 301, rev 59, p 28

SD-375, rev 8, p 16

Proposed References to be provided to applicants during examination: None

Learning Objective: 13.00.00.05 & 94.06.01.06 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AK3.01	
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor water level response

Proposed Question: RO Question # 45

The plant has been operating at 100% power with the reference leg backfill system out of service. These conditions have existed for the last 30 days.

Then the following sequence occurs:

- A SRV fails partially open and cannot be closed
- A manual scram is inserted
- Reactor pressure is 900 psig and lowering slowly
- As required by IPOI 5, Reactor Scram, Enhanced Reactor Vessel Level Monitoring is initiated IAW OI 880, Non-Nuclear Instrumentation System.
- As RPV pressure continues to lower, RPV level perturbations are noted on various level indicators.

Based on the above:

Water is being displaced in the reference leg by gases coming out of solution, causing indicated level to ____ (1) ____. The condensing chamber then refills the reference leg resulting in a return to actual level.

Under these conditions the _________ (2) _______ should be used to determine actual vessel level.

A. (1) INCREASE
(2) Narrow Range Gemacs

B. (1) DECREASE
(2) Narrow Range Gemacs

C. (1) INCREASE
(2) Wide Range Yarways

D. (1) DECREASE
(2) Wide Range Yarways

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The Wide Range Yarways are to be used. Plausible in that non-condensable gas build up occurs in the “GEMAC” reference leg. However the Narrow Range Gemacs also use this reference leg.

B. Incorrect: Water displacement in the reference leg causes an increase in indicated level. Additionally, the Wide Range Yarways are to be used.

C. Correct: With the Reference Leg Back Fill System out of service, gases will accumulate in the reference legs. As gases come out of solution in the reference leg as the vessel depressurizes, water is displaced. This results in a lowering differential pressure between the reference leg and the variable leg. The lowering differential pressure results in an indicated level rise. The condensing pot then restores the level of the reference leg resulting in a return to actual level.

IAW Section 6.1 of OI 880 (J-1), page 17, the Wide Range Yarways are to be used to determine actual level.

D. Incorrect: Water displacement in the reference leg causes an increase in indicated level.

Technical Reference(s): SD 880, page 20 (Attach if not previously provided)
Section 6.1 of OI 880 (J-1), page 17

Proposed References to be provided to applicants during examination: None

Learning Objective: 88.00.00.05 (As available)

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

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 5
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK3.05	
	Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements: Plant-Specific

Proposed Question: RO Question # 46

The plant is operating in single loop. The running Recirc Pump speed is 45%.

(1) Which one of the following instruments is used to determine core flow?

AND

(2) Why is this instrument used?

A. (1) 1C05 Total Core Flow recorder FR-4635.
(2) Because there is reverse flow in the idle jet pumps at this pump speed.

B. (1) 1C05 Core Plate Differential Pressure recorder PDR/FR-4528.
(2) Because there is forward flow in the idle jet pumps at this pump speed.

C. (1) 1C38 Jet Pump Total Head indication PDI-4567.
(2) Because there is forward flow in the idle jet pumps at this pump speed.

D. (1) 1C05 Core Plate Differential Pressure recorder PDR/FR-4528.
(2) Because there is reverse flow in the idle jet pumps at this pump speed.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: At recirc pump speeds < 50% flow shifts in the idle jet pumps from reverse to forward and the Total Core Flow indication becomes inaccurate because the loop flow summers subtract the idle loop flow from the operating loop when in single loop. Plausible if the candidate believes this happens at a lower pump speed because then there would still be reverse flow in the idle loop.

B. Correct: The Total core flow indication becomes inaccurate at running pump speeds < 50% because the loop flow summers subtract the idle loop flow from the operating loop when in single loop. However at low running pump speeds the flow in the idle loop jet pumps shifts from reverse to forward flow. At speed less than 50% procedures direct that core plate delta P indications be used to determine core flow.

C. Incorrect: Core plate dp indication is used at speeds < 50%. Plausible because this indication could be converted into a core flow determination if the correlation was provided to the operator.

D. Incorrect: At this pump speed there is forward flow in the idle jet pumps. Plausible if the candidate recalls that a different indication must be used at this pump speed but does not recall why.

Technical Reference(s): AOP 255.2, Note on page 4 (Attach if not previously provided)

OI 264, section 6.1, 1st two notes.

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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

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

Comments:

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

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Drywell venting

Proposed Question: RO Question # 47

Primary containment pressure is rising following a large RCS break inside the primary containment. Primary containment parameters are as follows:

- Drywell Pressure indication is 49 psig and slowly rising
- Torus pressure indication is 48 psig and slowly rising
- Narrow Range Torus level indication is pegged high at 16 feet

IAW EOP 2, Primary Containment Control, which ONE of the following is correct regarding the vent path that should be used and why?

A. 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.

The filtering of the SBGT system will reduce the offsite release rate.

B. 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.

This is the ONLY vent path available due to the level in the Torus.

C. The hardened vent path via CV-4300, INBD TORUS VENT ISOL and CV-4357, HARD PIPE VENT.

This path eliminates the potential for duct work or SBGT failure during venting which would significantly increase radioactivity levels in the reactor building.

D. 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.

Drywell venting is always preferred while in the EOPs because it has the most immediate and direct effect on Drywell pressure.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Torus vent paths are not allowed when torus water level is above 16 feet. Plausible in that torus venting is normally preferred due to the scrubbing provided by the torus and this path also utilizes SBGT.

B. Correct: With torus level above 16 feet the drywell must be vented because the torus vent paths are now flooded or at least must be assumed to be flooded.

C. Incorrect: This vent path is also via the torus and Torus vent paths are not allowed when Narrow Range torus water level is above 16 feet.

D. Incorrect: If available, torus venting is preferred. The reason why the drywell vent is used in this case is because the level in the torus precludes the use of torus vent paths. Plausible in that venting the drywell will have more of an immediate effect on drywell pressure than venting the torus but is still normally less preferred than the torus venting because of the scrubbing by the Torus.

Technical Reference(s): EOP-2 step PC/P-10 (Attach if not previously provided)
LP 95.59, Primary Containment Control, page 15.

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95.59, Objective 95.64.16.01 (As available)

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

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 9
55.43

Shielding, isolation, and containment design features, including access limitations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EA1.03	
	Importance Rating	3.9	

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

Proposed Question: RO Question # 48

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 sequence of events that occurred is as follows:

- T= 0 minutes : EOP 2 is entered when Drywell temperature exceeds 150°F
- T= 5 minutes: Drywell pressure exceeds 2 psig and the reactor scrams
- T= 15 minutes: Drywell spray was initiated to maintain Drywell temperature < 280°F

Note: RPV level lowered to 150 inches on the SCRAM and is now being maintained 170-211 with Condensate and Feed.

Assuming no other operator action has been taken, which ONE 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.
No other automatic action occurred.

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

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

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

Proposed Answer: C

Explanation (Optional):

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

Technical Reference(s): SD 760, page 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 573 - Primary Containment, Control, and Monitoring, objective 42.01.01.02 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

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

Comments:

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 # 49

The plant is operating at 100% power when a turbine trip occurs. Additional information is as follows:

- ”A” RPS trip system actuates as designed
- ”B” RPS trip system fails to automatically actuate
- Division 1 EOC-RPT actuates as designed
- Division 2 EOC-RPT fails to actuate

Two minutes later, “B” RPS is manually tripped.

Based on the above, what is the response of the Recirculation System?

A. Both Recirc Pumps trip.

B. “A” Recirc Pump trips.
“B” Recirc Pump runs back to minimum speed following control rod insertion.

C. Neither Recirc Pump trips.
Both Recirc Pumps run back to minimum speed following control rod insertion via the trip of “B” RPS.

D. Neither Recirc Pump trips.
Both Recirc Pumps run back to minimum speed following control rod insertion due to lowering feedwater flow.

Proposed Answer: A

Explanation (Optional):

A. Correct: Each division can perform the ATWS function. Each division controls one RPT for each Recirc Pump.

B. Incorrect: Both pumps trip. Plausible if the candidate thinks that Division 1 will only trip the “A” recirc pump but understands that only one ARI valve is required to insert the control rods. If so, the candidate would conclude that “B” Recirc pump would run back to minimum speed when feed flow lowers.

C. Incorrect: Both Recirc pumps trip. Plausible if the candidate believes that both divisions are required to trip the recirc pumps and open the ARI valves. If so, the candidate would conclude that the recirc pumps would run back to minimum speed when feed flow lowers following the RPS trip.

D. Incorrect: Both Recirc pumps trip. Plausible if the candidate believes that both divisions are required to trip the recirc pumps but only one division is required to open an ARI valve. If so, the candidate would conclude that both Recirc pumps would run back to minimum speed when feed flow lowers.

Technical Reference(s): SD 358, pages 24 - 26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 264, 12.00.00.02.C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AA1.06	
	Importance Rating	3.0	

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: RO Question # 50

The plant is operating at 100% power when annunciator 1C40A (F-3) MAIN GEN EXCITER SMOKE DETECTOR OR EXCITER CARDOX SYS INITIATED alarms in the control room.

Which ONE of the following describes the Cardox System response and action(s) required for this alarm condition? Assume that a fire exists in the Main Generator Exciter.

The Cardox System ___ (1) ___.

Following closure of the exciter vent exhaust dampers, the Main Turbine must be tripped ___ (2) ___.

A. (1) will automatically initiate
(2) immediately

B. (1) will automatically initiate
(2) within 2 minutes

C. (1) must be manually initiated
(2) immediately

D. (1) must be manually initiated
(2) within 2 minutes

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The Cardox requires manual initiation. Additionally IAW OI 513 the main turbine shall be tripped within 2 minutes of the closure of the exciter vent exhaust dampers.

B. Incorrect: The Cardox requires manual initiation.

C. Incorrect: IAW OI 513 the main turbine shall be tripped within 2 minutes of the closure of the exciter vent exhaust dampers.

D. Correct: IAW the ARP the system must be manually initiated. Additionally the turbine must be tripped within two minutes of damper closure.

Technical Reference(s): ARP 1C40A (F-3)
OI 513, P7L #15 and 17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 4
55.43
Secondary coolant and auxiliary systems that affect the facility

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #		
	K/A #	295037 EA2.06	
	Importance Rating	4.0	

Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure

Proposed Question: RO Question # 51

An ATWS is in progress following an inadvertent Group 1 isolation.

- RPV pressure is currently cycling between 1080 and 1135 psig
- Reactor power is cycling between 30% and 40%
- Standby Liquid is injecting

Based on the above and IAW ATWS – RPV Control, what action is now required for RPV pressure control?

A. Open additional SRVs as required and stabilize pressure below 880 psig.

B. Open additional SRVs as required and stabilize pressure below 1055 psig.

C. Open additional SRVs manually and commence a plant cooldown not to exceed 100°F per hour.

D. Operate SRVs manually while maintaining pressure between 1080 and 1135 psig. Alternate SRVs to help prevent localized Torus heatup.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The required action is to stabilize below 1055 psig. Plausible in that if the bypass valves were available EHC pressure set would be lowered to this pressure.

B. Correct: IAW EOP steps P2 through P-4, if SRVs are cycling the required action is to terminate the cycling by manually opening SRVs and stabilizing pressure below 1055 psig.

C. Incorrect: A plant cooldown is not authorized until the cold shutdown boron weight has been injected.

D. Incorrect: The required action is to terminate the SRV cycling by opening additional SRVs. Plausible in that ATWS-RPV control in general does not allow a pressure reduction/cooldown unless the reactor is shutdown. It also directs that if SRVs are being manually operated to alternate the SRVs being used to prevent localized heatup. However terminating the cycling of the SRVs takes priority.

Technical Reference(s): ATWS – RPV Control, steps P-2 and P-4. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 50007-95, LO 95.56.08.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295018	AA2.04	
	Importance Rating	2.9		

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

Proposed Question: RO Question # 52

RHR Loop “A” is aligned for Torus Cooling during accident conditions. RHRSW conditions are as follows:

- RHRSW pumps “A” and “C” are in service
- “A” RHRSW TO RHR ΔP indicator PDI-2046 is indicating 25 psid
- “A” RHRSW INLET FLOW indicator FI-2050 is indicating 5000 gpm

Then, RHRSW pump “A” trips and parameters change as follows:

- “A” RHRSW TO RHR ΔP indicator PDI-2046 lowers to 18 psid
- “A” RHRSW INLET FLOW indicator FI-2050 lowers to 2800 gpm

Based on the above, which ONE of the following is correct?

“C” RHRSW pump is…

A. exceeding runout limits.

B. exceeding cavitation limits.

C. within all flow limits, but the ΔP across the heat exchanger is too low.

D. within all flow limits, and the ΔP across the heat exchanger is also within limits.

Proposed Answer: A

Explanation (Optional):

A. Correct: Per precaution 15 of OI 416, RHRSW pump flow must be limited to < 2600 gpm to prevent runout.

B. Incorrect: Per precaution 15 of OI 416, RHRSW pump flow must be above 1200 gpm to prevent cavitation.

C. Incorrect: The RHRSW pump is exceeding runout limitations.

D. Incorrect: The RHRSW pump is exceeding runout limitations. Additionally the ΔP across the heat exchanger must be above 20 psid (precaution #3).

Technical Reference(s): OI 416 RHR SERVICE WATER SYSTEM, pages 3 and 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 416, objective 30.01.01.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA2.01	
	Importance Rating	3.6	

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Area radiation levels

Proposed Question: RO Question # 53

The plant is operating at 100% power with spent fuel being moved in the spent fuel pool when the control room is notified that the spent fuel pool water level is lowering.

Additional conditions are as follows:

- SPENT FUEL STORAGE AREA ARM HI RAD annunciator has alarmed. RI-9178 is indicating 500 mR/hr and rising slowly.
- REFUELING FLOOR NORTH END HI RADIATION annunciator has alarmed. RI-9163 is pegged high at > 100 mR/hr.
- REFUELING FLOOR SOUTH END HI RADIATION annunciator has alarmed. RI-9164 is pegged high at > 100 mR/hr.
- CONTROL BLDG INTAKE AIR RAD MON RIM-6101A and B HI/TROUBLE annunciators have both alarmed. RIM-6101A and B are both reading 5 mR/hr and rising slowly.
- “A” and “B” Control Building Standby Filter Units are in standby readiness.
- EOP 3, Secondary Containment Control, has been entered.

Based on the above:

IAW EOP 3, entry into EOP 1 RPV Control and its associated manual scram is ___ (1) ____.

Manual starting of the Control Building Standby Filter Unit is ___ (2) ____.

A. (1) required
(2) required

B. (1) required
(2) NOT yet required

C. (1) NOT yet required
(2) required

D. (1) NOT yet required
(2) NOT yet required

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Entry into EOP 1 is Not required. EOP 3 directs that EOP 1 be entered if a pressure reduction will reduce the leak rate (steps SC 4 and 5). Given that the conditions describe a spent fuel pool leak, this is not the case. Plausible in that EOP 3 also asks whether a parameter has exceeded its Max Safe Value, and if the leaking system is a primary system then EOP 1 entry is required. Two ARMs have exceeded their Max Safe Values in that they are pegged high.

B. Incorrect: Entry into EOP 1 is Not required. Also a manual start of the Control Building Standby Filter Unit (SFU) is required. The Control Building SFUs are designed to auto start when RIM-6101A and B exceed 3.5 mR/hr. Since they failed to auto start, they are required to be manually started.

C. Correct: Entry into EOP 1 is Not required as described above. Also manual starting of the SFU is required.

D. Incorrect: Manual starting of the SFU is required.

Technical Reference(s): EOP 3 (Attach if not previously provided)

ARP 1C26A[B], C-2

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95.68, objective 95.70.07.02 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 2.1.1	
	Importance Rating	3.8	

Conduct of Operations: Knowledge of conduct of operations requirements (Partial or Complete Loss of Forced Core Flow Circulation)

Proposed Question: RO Question # 54

The plant was operating at 15% power when an inadvertent closing of the "A" Recirc Pump discharge valve occurs. The CRS has directed the ATC operator to attempt to open the discharge valve. The BOP operator has gone to the EHC controls to adjust the EHC setpoint. Which ONE of the following applies to these conditions?

A. The simultaneous performance of these activities is permitted because the "A" Recirc Pump has tripped.

B. The simultaneous performance of these activities is permitted because reactor pressure will NOT be affected.

C. The simultaneous performance of these activities is NOT permitted because of the potential for a reactor scram.

D. The simultaneous performance of these activities is NOT permitted because of the possible addition of positive reactivity.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Performance of these activities is NOT permitted because of the possible addition of positive reactivity from two sources. Plausible because the candidate may assume closing the discharge valve in this condition would trip the pump. However at this power the recirc pumps will be on minimum speed and no pump trips will occur.

B. Incorrect: Performance of these activities is NOT permitted because of the possible addition of positive reactivity from two sources. Plausible because the candidate may believe that the actions taken will NOT increase reactor power, however any closure of the bypass valves will cause a positive reactivity addition.

C. Incorrect: Because the reactor mode switch is in RUN at this power level actions performed will NOT cause a reactor scram.

D. Correct – Opening the recirc pump discharge valve will raise recirculation flow, while operating the EHC controls may affect bypass valve position affecting reactor pressure and also adding positive reactivity. IAW OP-AA-103-1000 Reactivity Control, positive reactivity additions shall not be made via simultaneous use of control rods and dilution on lowering Tavg for PWRs or via simultaneous use of control rods, recirc flow, or EHC Pressure for BWRs.

Technical Reference(s): OP-AA-103-1000 pg.13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 2.1.20	
	Importance Rating	2.7	

Ability to interpret and execute procedure steps: Partial or Complete Loss of AC

Proposed Question: RO Question # 55

The plant is operating at 100% power with the electric plant configured for normal full power operation.

Then, the following alarms and indications are received:

- 1C08A (A-6), S/U XFMR TO 1A3 BREAKER 1A302 TRIP
- 1C08A (A-5), BUS 1A3 LOCKOUT TRIP
- 1C08A (C-6), BUS 1A3 LOSS OF VOLTAGE
- Breaker 1A301, STANDBY TRANSFORMER TO BUS 1A3, indicates open
- Breaker 1A302, STARTUP TRANSFORMER TO BUS 1A3, indicates open
- Breaker 1A311, A DIESEL GENERATOR 1G-31, indicates open
- “A” SBDG 1G-31 is running at rated voltage and frequency

Given the above information, which ONE of the following immediate actions is required by AOP 301 and why?

A. Manually close ”A” SBDG output breaker, 1A311. The breaker should have closed.

B. Manually close the Standby Transformer to bus 1A3, 1A301. A fast transfer should have occurred.

C. Manually close the Standby Transformer to bus 1A3, 1A301. A slow transfer should have occurred.

D. Manually trip the “A” SBDG by placing its control switch in pull-to-lock. The diesel is running without cooling water.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The BUS 1A3 LOCKOUT TRIP alarm indicates that the bus lockout relay tripped LOR 186-3 due to a bus fault. This relay prevents ‘A’ SBDG output breaker, 1A311 from auto closing.

B. Incorrect: The BUS 1A3 LOCKOUT TRIP alarm indicates that the bus lockout relay tripped LOR 186-3 due to a bus fault. This relay prevents Standby Transformer to bus 1A3, 1A301 from auto closing.

C. Incorrect: The BUS 1A3 LOCKOUT TRIP alarm indicates that the bus lockout relay tripped LOR 186-3 due to a bus fault. This relay prevents Standby Transformer to bus 1A3, 1A301 from auto closing.

D. Correct: The BUS 1A3 LOCKOUT TRIP alarm indicates that the bus lockout relay tripped LOR 186-3 due to a bus fault. The automatic actions associated with this relay tripping are:

Breakers 1A301 and 1A302 trip open and are interlocked from manually or automatically closing
Bus 1A3 load sheds.
"A" SBDG 1G-31 auto starts on Bus 1A3 undervoltage, then runs up to speed and frequency.
Breaker 1A311 does NOT auto close and cannot be manually closed.
With the diesel running with a bus lockout, there is no ESW to the diesel. AOP 301 directs that the diesel be shutdown by placing the HS3231A in pull-to-lock.

Technical Reference(s): ARP 1C08A A-5 (Attach if not previously provided)
ARP 1C08A A-6
AOP 301, Immediate Action

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 304.1, 15.00.00.03 (As available)

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

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 2.4.45	
	Importance Rating	4.1	

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm: High Off-site Release Rate

Proposed Question: RO Question # 56

The plant is operating at 100% power with plant ventilation systems in a normal full power configuration.

Then the plant scrams following a plant transient. Both SBGT trains start and a complete Group 3 isolation occurs. The following indications are then noted:

- Alarm REACTOR BUILDING KAMAN 3, 4, 5, 6, 7 & 8 RAD TROUBLE (IC35A C-3) annunciates.
- The alarm is determined to be due to rising radiation levels on the KAMAN monitors.

IAW with ARP IC35A C-3 and OI 170, STANDBY GAS TREATMENT SYSTEM, which ONE of the following identifies the earliest point where operator action is required?

When the KAMAN monitors reach the…

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

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

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

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

Proposed Answer: A

Explanation (Optional):

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

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

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

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

Technical Reference(s): ARP IC35A C-3 and OI 170 precaution #10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 879.3, objective 87.01.01.01 (As available)

Question Source: Bank # DAEC # 47017

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AA2.02	
	Importance Rating	3.5	

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

K/A Justification: The DAEC generator capability curve is a function of Megavar loading and Megawatt electric loading. Main Generator voltage is not plotted on this curve. However there is a direct correlation between grid voltage and the amount of Megavar loading on the DAEC generator and therefore on generator capability curve limits.

Proposed Question: RO Question # 57

The plant is operating at 100% power with the Main Generator in its normal full power configuration, when ITC MIDWEST notifies DAEC of potential grid instabilities over the next few hours.

Current Main Generator conditions are as follows:

- 640 MWe Gross
- 150 MVARS, lagging (out)
- 45 psig Hydrogen pressure

One hour later grid voltage is observed to be slowly lowering.

If uncorrected, the lowering grid voltage will result in the Main Generator exceeding the Main Generator Estimated Capability Curve when…

A. MVAR loading exceeds -240 MVAR leading.

B. MVAR loading exceeds 260 MVAR lagging.

C. MVAR loading reaches 210 MVAR lagging based on exceeding the rated PF.

D. MVAR loading reaches -100 MVAR leading based on exceeding the URAL limit.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: DAEC MVAR loading increases in the Lagging direction as grid voltage lowers relative to main generator terminal voltage. Plausible if the candidate is unsure of how MVAR loading varies with changes in grid voltage. (For 640 MWe and 45 psig H2, the curve is exceeded in the leading direction at -240 MVARs)

B. Correct: As grid voltage lowers, the DAEC generator will tend to "pickup" additional reactive loading. Using the Estimated Capability Curve, the curve will be exceeded when MVAR loading increase to 260 MVAR, Lagging.

C. Incorrect: The curve will not be exceeded until MVAR loading increase to 260 MVAR, Lagging. 210 MVAR is when the .95 PF (power factor) line is exceeded for a real loading of 640 MWe. Plausible if the candidate believes the .95 PF (power factor) is a limit. This is plausible in that the generator ratings are based on a .95 PF – see the ratings at the top of the curve. However the individual power factors are not limits but merely define the relationship between MWe and MVAR loading.

D. Incorrect: DAEC MVAR loading increases in the Lagging direction as grid voltage lowers relative to main generator terminal voltage. The URAL limit is only applicable when MVAR loading is in the Lead direction. Plausible if the candidate is unsure of how MVAR loading varies with changes in grid voltage.

Technical Reference(s): OI 698, Appendix 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OI 698, Appendix 1

Learning Objective: LP 698, 57.00.00.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK3.03	
	Importance Rating	3.5	

Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Disabling control room controls

Proposed Question: RO Question # 58

The control room has been evacuated due to a fire in the main control room.

Regarding RHR operation via the Remote Shutdown System:

What is the reason for placing the RHR related Remote Shutdown System transfer switches in the EMER position following a control room evacuation?

These switches …

A. install separate auto isolation circuits to ensure that fire damage will NOT prevent an automatic isolation on low vessel level.

B. install separate auto initiation circuits to ensure that fire damage will NOT prevent an automatic LPCI injection on Low-Low-Low vessel level.

C. ensure that fire induced circuit faults associated with normal control circuits are isolated and will NOT prevent RHR operation from the Remote Shutdown panels.

D. ensure that fire induced faults in control power circuits will NOT prevent RHR operation by transferring RHR pump and valve power supplies to alternate power supplies.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The low level isolation for SDC is bypassed when the transfer switches are in EMER.

B. Incorrect: LPCI initiation signals and LPCI valve logics are bypassed when the transfer switches are in EMER.

C. Correct: The design basis scenario for the adequacy of the Remote Shutdown Panel System assumes that normal control circuits may either short circuit or open circuit due to the fire. Therefore, to take credit for having equipment available, fire damaged circuits must be isolated and replaced by non-damaged control circuits. This is the function of the transfer switches.

D. Incorrect: Power to the pumps and valves is not transferred. Plausible in that the "yellow" transfer switches will transfer components to the emergency fuses in case a normal control circuit has blown due to fire damage. However the power supply remains the same.

Technical Reference(s): SD 925, page 24 (Attach if not previously provided)
SD 925, Functional description of transfer switch operation on REMOTE SHUTDOWN PANEL 1C-388 (page 10)

Proposed References to be provided to applicants during examination: None

Learning Objective: STG AOP 915, 94.28.06.02 (As available)

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

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 7
55.43

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

Comments:

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

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

Proposed Question: RO Question # 59

During ATWS conditions Boron was injected and RPV level intentionally lowered to reduce reactor power.

RPV level is now being controlled between -25 inches and +15 inches.

Based on the above:

(1) When can RPV level FIRST be raised to the normal control band of +170 inches to +211 inches

AND

(2) Why is water level raised at this time?

A. (1) When the Hot Shutdown Boron Weight is injected
(2) So that a controlled plant cooldown can be commenced without challenging adequate core cooling.

B. (1) When the Cold Shutdown Boron Weight is injected
(2) So that a controlled plant cooldown can be commenced without challenging adequate core cooling.

C. (1) When the Hot Shutdown Boron Weight is injected
(2) To mix the Boron that has accumulated below the core plate with the water in the core by increasing natural circulation flow.

D. (1) When the Cold Shutdown Boron Weight is injected
(2) To mix the Boron that has accumulated below the core plate with the water in the core by increasing natural circulation flow.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Level is raised to mix the boron that may have accumulated in the lower plenum. Plausible in that a plant cooldown is contingent upon how much boron has been injected.

B. Incorrect: Level is raised to mix the boron that may have accumulated in the lower plenum. Additionally level is raised as soon as the hot shutdown boron weight is injected.

C. Correct: IAW with the ATWS EOP, steps L-7 and L-8, level is raised when the Hot Shutdown Boron Weight is injected. When level was lowered to below TAF, little if any natural circulation was occurring. The boron that was being injecting was accumulating in the lower plenum due to the low flow conditions. Once the Hot Shutdown Boron Weight has been injected it is necessary to raise level and re-establish natural circulation in order to mix the boron.

D. Incorrect: IAW with the ATWS EOP, steps L-7 and L-8, level is raised when the Hot Shutdown Boron Weight is injected. Plausible in that other EOP actions are contingent on the Cold Shutdown Boron Weight Being injected.

Technical Reference(s): ATWS EOP, steps L-7 and L-8 (Attach if not previously provided)
ATWS Bases document

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95-50, EO 95.51.01.01 (As available)

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 10
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295014 AK2.01	
	Importance Rating	3.9	

Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: RPS

Proposed Question: RO Question # 60

With the plant operating at 100% power, CV-4416, “B” OUTBOARD MSIV and CV-4419, “C” OUTBOARD MSIV fail SHUT.

Which ONE of the following will be the impact of this failure?

This condition will result in…

A. NO automatic actions.

B. a half scram initiated by the "B" and "C" MSIV position.

C. a full scram initiated by reactor pressure.

D. a full scram initiated by the "B" and "C" MSIV position.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Plausible; would be true for an initial low power condition. With Mode Switch in STARTUP, no automatic actions occur. 75% is the normal power limitation with 3 Main Steam Lines.

B. Incorrect: Plausible; would be true for A and B, C and D, A and C, or B and D with a lower initial power. B and C MSIV do NOT cause a Half Scram based on valve positions.

C. Correct: Closure of one MSIV at full power raises reactor pressure to the alarm point, and power >120%. (Steam flow in the remaining lines also rises to near the MSL Hi Flow Group 1 isolation setpoint.). The sudden closure described causes transient power and pressure above the scram setpoints. This transient results in a full scram. It is required to enter IPOI-5 and place the Reactor Mode Switch in SHUTDOWN.

D. Incorrect: Plausible because closure of the two valves does cause a scram and MSIV position does cause a scram, however B and C MSIV do NOT cause a Full Scram based on valve positions, they would cause a half scram.

Technical Reference(s):	SD 358, pg 20 50000_683-0_lp page 26, 27, 34 Validated in DAEC Simulator 1/9/12		(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 358, 22.02.01.03 (As available)

Question Source:	Bank #	DAEC Bank # 47907	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content:	55.41	6
	55.43	

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295015 AK3.01	
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM: Bypassing rod insertion blocks

Proposed Question: RO Question # 61

Following a reactor scram, 6 control rods failed to fully insert.

The CRS has authorized the use of RIP 103.3, MANUALLY DRIVE CONTROL RODS, to complete the control rod insertion.

Which ONE of the following actions is directed by RIP 103.3 in order to prevent control rod blocks from interfering with manually inserting the rods?

A. Bypass the RWM to prevent any RWM insert blocks from interfering with rod insertion.

B. Bypass the RWM to prevent any RPIS Inoperative select blocks from interfering with rod insertion.

C. Bypass the SRM closest to the control rods being inserted to prevent SRM associated rod blocks.

D. Insert rods using ONLY the EMER ROD IN/NOTCH OVERRIDE switch because this will bypass all insert blocks.

Proposed Answer: A

Explanation (Optional):

A. Correct: RIP 103.3, step 3, directs that the RWM be bypassed. This is to prevent any RWM insert blocks from preventing rod insertion. Following a scram multiple insert errors would be indicated when one of the problem rods was selected for insertion resulting in a RWM insert block.

B. Incorrect: The RWM is bypassed in order to prevent insert blocks. An RPIS Inop condition would still result in a select block preventing rod insertion regardless of the position of the RWM bypass switch.

C. Incorrect: Bypassing an SRM will not prevent a RWM insert block. Plausible in that SRM reading near the control rods will change, however with the SRMs inserted lowering counts will NOT cause a rod block.

D. Incorrect: The EMER ROD IN/NOTCH OVERRIDE switch will not prevent a RWM insert block. Plausible in that use of this switch is directed in RIP 103.3 and it also acts directly on the directional control valves of the HCUs, bypassing the timer circuits.

Technical Reference(s): LP 95-03, page 17 (Attach if not previously provided)

RIP 103.3

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95-03, EO 95.09.01.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295008 AA1.02	
	Importance Rating	3.3	

Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: Reactor water cleanup (ability to drain): Plant-Specific

Proposed Question: RO Question # 62

Following a reactor scram, Reactor Water Cleanup (RWCU) is being used to assist in RPV level control. Additional information is as follows:

- RPV pressure is 920 psig, maintained via turbine bypass valves
- RPV level is 205 inches and slowly rising
- RWCU is currently draining to the main condenser IAW OI 261, section 7.1 RWCU SYSTEM VESSEL DRAIN WITH A RWCU PUMP IN OPERATION
- MO 2731, CLEANUP DRAIN TO MN COND is open
- Current drain flow to the Main Condenser is 40 gpm

Based on the above and IAW OI 261, which of the following actions is authorized in order to raise the drain flow rate and stabilize RPV level?

Action 1: Open CV-2729, Cleanup System Drain Header Control Valve further, using HC-2729 DRAIN FLOW REGULATOR

Action 2: Open MO-2727, DRAIN FLOW ORIFICE BYP

Action 3: Open MO-2732, CLEANUP DRAIN TO RW

A. Action 1 ONLY

B. Action 1 or 2 ONLY

C. Action 1 or 3 ONLY

D. Action 1, 2 or 3

Proposed Answer: A

Explanation (Optional):

A. Correct: Given the current conditions only action 1 is authorized

B. Incorrect: Per the caution in OI 261, section 7.1, when reactor pressure is above 600 psig, MO-2727 DRAIN FLOW ORIFICE BYP valve should be closed (protects low pressure piping downstream).

C. Incorrect: Per precaution 7 of OI 261, MO-2731 CLEANUP DRAIN TO MN COND valve and MO-2732 CLEANUP DRAIN TO RW valve should not be opened at the same time when there is a vacuum on the condenser. This can provide a path through Radwaste with a resultant loss of condenser vacuum. Since the bypass valve is open, there is a vacuum in the main condenser.

D. Incorrect: Opening MO-2727 or MO-2732 is not authorized.

Technical Reference(s): OI 261, section 7.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 261 RO objective 11.01.01.06 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295035 EA2.02	
	Importance Rating	2.8	

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Off-site release rate: Plant-Specific

Proposed Question: RO Question # 63

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

- A steam line break in the steam tunnel which could NOT be isolated has caused a reactor scram
- Fuel failures have occurred
- RPV pressure is 420 psig and lowering
- The steam tunnel blowout panels have released
- Steam tunnel temperatures have exceed their max safe values
- The main steam line radiation levels have exceed their Hi-Hi setpoints
- An ALERT has just been declared based on Offsite Rad levels

(1) Which ONE of the following is the most significant release pathway?

AND

(2) When is emergency depressurization required?

A. (1) Steam tunnel → Turbine building → Turbine building exhaust fans → environment.
(2) Emergency depressurization is currently required

B. (1) Steam tunnel → Turbine building → Turbine building exhaust fans → environment.
(2) Before offsite release rates exceed that level requiring a General Emergency

C. (1) Steam tunnel → Reactor Building Vent Shaft→ Standby Gas Treatment → Offgas stack → environment.
(2) Emergency depressurization is currently required

D. (1) Steam tunnel → Reactor Building Vent Shaft→ Standby Gas Treatment → Offgas stack → environment.
(2) Before offsite release rates exceed that level requiring a General Emergency

Proposed Answer: B

Explanation (Optional):

A. Incorrect: An entry condition to EOP 4 was exceeded when offsite release rates exceeded that requiring an Alert. EOP 4 requires an emergency depressurization be performed before release rates exceed that requiring a General Emergency. Although EOP 3 entry conditions have also been exceeded there are no conditions currently requiring an ED since the max safe temperature is limited to one area.

B. Correct: A blowout panel in the steam tunnel will vent the steam tunnel to the turbine building when pressure in the steam tunnel exceeds 7 inches of water. An entry condition to EOP 4 was exceeded when offsite release rates exceeded that requiring an Alert. EOP 4 requires an emergency depressurization be performed before release rates exceed that requiring a General Emergency.

C. Incorrect: A blowout panel in the steam tunnel will vent the steam tunnel to the turbine building when pressure in the steam tunnel exceeds 7 inches of water. Plausible if the candidate does not recognize that the blowout panel will vent the steam tunnel as this would then be the correct flow path. Also there are no conditions currently requiring an ED.

D. Incorrect: A blowout panel in the steam tunnel will vent the steam tunnel to the turbine building when pressure in the steam tunnel exceeds 7 inches of water.

Technical Reference(s): SD 170-1, page 7 (Attach if not previously provided)

EOP 4, step RR-4

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95.71, objective 95.71.01.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 13

55.43

Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295034 2.4.2	
	Importance Rating	4.5	

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions: Secondary Containment Ventilation High Radiation

Proposed Question: RO Question # 64

The plant is shut down and cooling down for a forced outage. A Drywell air purge has been established using "A" SBGT train.

Then, both RB Vent Shaft Rad Monitors, RIM 7606A and B, rise to the EOP 3, Secondary Containment Control entry condition.

Which ONE of the following is correct regarding the impact on SBGT and the drywell air purge lineup?

"B" SBGT will ….

A. not automatically start but should be started manually.
The drywell air purge valves will NOT isolate.

B. not automatically start but should be started manually.
The drywell air purge Valves will isolate.

C. automatically start.
The drywell air purge valves will NOT isolate.

D. automatically start.
The drywell air purge valves will isolate.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The EOP 3 entry condition on RB Vent Shaft Rad high radiation is also the setpoint for the Group 3 isolation. Both radiation monitors reaching this setpoint will result in a Channel 'A' and Channel 'B' trips within the Group 3 logic. This will result in the 'B' train of SBGT starting and drywell air purge lineup isolating. Plausible if the candidate is not aware of the correlation between the entry condition and the Group 3 setpoints.

B. Incorrect: Both radiation monitors reaching this setpoint will result in a Channel ‘A’ and Channel ‘B’ trips within the Group 3 logic. This will result in the ‘B’ train of SBGT starting and drywell air purge lineup isolating.

C. Incorrect: Both radiation monitors reaching this setpoint will result in a Channel ‘A’ and Channel ‘B’ trips within the Group 3 logic. This will result in SBGT starting and drywell air purge lineup isolating.

D. Correct: The EOP 3 entry condition on RB Vent Shaft Rad high radiation is also the setpoint for the Group 3 isolation. Both radiation monitors reaching this setpoint will result in a Channel ‘A’ and Channel ‘B’ trips within the Group 3 logic. This will result in the ‘B’ train of SBGT starting and drywell air purge lineup isolating.

Technical Reference(s): ARP 1C05B, C-8 (Attach if not previously provided)

EOP 3 entry conditions

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 95.68, objective 95.00.00.22 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295020 AK2.12	
	Importance Rating	3.1	

Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Instrument air/nitrogen

Proposed Question: RO Question # 65

The plant is operating at 100% power with the Drywell pneumatic system aligned for normal full power operation.

Then, an inadvertent Group 3 isolation occurs.

Which ONE of the following is correct regarding the impact of the isolation on the Drywell pneumatics and Drywell components?

A. The Drywell pneumatic supply line will isolate. Local accumulators inside the Drywell will provide for 5 actuations of the inboard MSIVs.

B. The Drywell pneumatic supply line will isolate. Local accumulators inside the Drywell will provide for 5 actuations of the ADS valves until the isolation can be corrected.

C. The Drywell N2 compressor will trip when Compressor Suction Isolation Valves, CV-4378A & CV-4378B close. Emergency N2 Backup Supply Valve CV-4377 will open to maintain the pneumatic supply to the Drywell.

D. The Drywell N2 compressor will trip when Compressor Suction Isolation Valves, CV-4378A & CV-4378B close. Nitrogen Accumulator 1T128 will maintain the pneumatic supply to the Drywell until the isolation can be corrected.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The MSIVs are not assured of remaining open for this period of time. Plausible in that there are accumulators associated with the MSIVs. However the accumulator volume is adequate to provide full stroking of the valve, (NOT five) for only one half cycle (open to close) when supply gas to the accumulator has failed.

B. Correct: In addition to the compressor suction valves, compressor isolation valve, CV-4371A also closes, isolating the pneumatic supply line. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that following a failure of the pneumatic supply to the accumulator; each ADS valve can be actuated at least 5 times up to 100 days following a LOCA.

C. Incorrect: In addition to the compressor suction valves, compressor isolation valve, CV-4371A also closes, isolating the pneumatic supply line. Although Emergency N2 Backup Supply Valve CV-4377 will open as accumulator pressure lowers, the drywell will remain isolated.

D. Incorrect: In addition to the compressor suction valves, compressor isolation valve, CV-4371A also closes, isolating the pneumatic supply line. Accumulator 1T128 is upstream of these isolation valves and will have no bearing on maintaining drywell pneumatics.

Technical Reference(s): SD 183-1, pages 7 -12 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G1 2.1.17	
	Importance Rating	3.9	

Conduct of Operations: Ability to make accurate, clear and concise verbal reports.

Proposed Question: RO Question # 66

Which ONE of the following statements describes the annunciator reporting requirements after the CRS has announced entry into abnormal or emergency operating procedures?

A. Do NOT announce any alarms unless directed by the CRS.

B. ALL alarms shall be announced unless given specific instructions from the CRS.

C. Announce ONLY alarms which represent Emergency Operating Procedure entry conditions.

D. Announce ONLY those significant alarms needed to implement Emergency Operating Procedures and Abnormal Operating Procedures.

Proposed Answer: D

Explanation (Optional):

A. Incorrect –Because during EOP operations there are too many alarms and alarm prioritization can be difficult, it is plausible the CRS could direct which alarms they wish to monitor.

B. Incorrect - This is plausible because the normal alarm response requires announcing ALL unexpected alarms.

C. Incorrect – This is plausible because these alarms are announced, however all annunciators significant to implementing EOPs and AOPs must be announced.

D. Correct - Under transient conditions, the CRS will announce entry into abnormal or emergency operating procedures. The operators are then allowed to announce only those significant alarms needed to implement those procedures.

Technical Reference(s): OP-AA-100-1000, Att. 1, step 9, pg 13-14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:			(As available)
Question Source:	Bank #	DAEC Bank # 47329(1412)	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:		Last NRC Exam:	Not Used
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.20	
	Importance Rating	4.6		

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 67

A control room operator is running a Standby Diesel Generator Surveillance Test Procedure (STP). After the Operator completes each step of the surveillance, the Operator places their initials beside the step.

While conducting the test, an unexpected alarm is received in the Control Room. The Operator acknowledges and announces the alarm, ensures that the Diesel Generator is in a stable condition, flags the current step in the STP, lays the STP to the side, and pulls out the Annunciator Response Procedure (ARP) for the alarm.

The operator reviews the steps in the ARP, sets it down, and without any further reference to the ARP, begins taking the stated actions.

The Operator then gets permission from shift supervision and returns to the STP flagged step, performs a Job Site Review, and continues on with the STP performance.

Which ONE of the following is correct regarding the operator’s compliance with the requirements for procedural adherence?

A. NO, the operator has NOT complied. STPs cannot be suspended in order to take other actions.

B. YES, the operator has complied. STPs are continuous use procedures, and ARPs are reference use procedures.

C. NO, the operator has NOT complied. The ARP is a reference use procedure and must be either place kept, or periodically reviewed and reviewed upon completion.

D. YES, the operator has complied. Operators are allowed to take any actions deemed necessary to respond to the alarm even if it conflicts with an approved procedure (STP).

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The procedure may be exited procedure if it will be resumed in a timely manner. Plausible if the candidate believes exiting the procedure requires exiting the procedure.

B. Incorrect – Reference use procedures require place keeping and reviewing the actions taken. Plausible because the candidate may not realize the actions required for a reference use procedure and literally treat the procedure as a reference.

C. Correct – The ARP is a “Reference Use” procedure, requiring the CRO to complete each step in the sequence specified unless the procedure specifically allows otherwise. Then mark off steps when completed and review the procedure upon completion to verify required actions were completed.

D. Incorrect - Reference use procedures require place keeping and reviewing the actions taken. Plausible because the candidate may not realize the actions required for a reference use procedure and literally treat the procedure as a reference.

Technical Reference(s): AD-AA-100-1006, Sect 4.2, pgs 13-15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 19661(1412)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.39	
	Importance Rating	3.9	

Equipment Control: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question: RO Question # 68

The plant is operating at 100% power when the CRS determines the “A” Standby Diesel Generator (SBDG) is inoperable.

Which ONE of the following is required by Technical Specifications?

Within 1 hour…

A. determine that the Offsite Circuits are operable.

B. determine that the “B” SBDG is not inoperable due to common cause failure.

C. demonstrate that the “B” SBDG is operable by running its operability surveillance test.

D. declare the required feature(s) supported by the ”A” SBDG inoperable when redundant required feature(s) are inoperable.

Proposed Answer: A

Explanation (Optional):

A. Correct. – With one DG inoperable T.S. 3.8.1.B.1 requires performing SR 3.8.1.1 for OPERABLE offsite circuit(s). This STP verifies correct breaker alignment and indicated power availability for each offsite circuit capable of supplying the onsite Class 1E AC Electrical Distribution System.

B. Incorrect. – T.S. 3.8.1.B.3, requires determining the OPERABLE DG is not inoperable due to common cause failure.

C. Incorrect. – T.S. 3.8.1.B.3, requires performing SR 3.8.1.2 for the operable DG once per 72 hours.

D. Incorrect. – T.S. 3.8.1.B.2, requires declaring the required features(s) supported by inoperable DG inoperable when redundant required feature(s) are inoperable within 4 hours of the discovery of the SBDG being inoperable.

Technical Reference(s): TS 3.8.1
STP 3.8.1-01, pg 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: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.22	
	Importance Rating	4.0		

Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: RO Question # 69

Which ONE of the following constitutes a Technical Specifications safety limit violation?

A. Reactor Water level is currently 25 inches above TAF.

B. Core flow is 20% of rated and thermal power is ≤21.7% RTP.

C. Reactor steam dome pressure reaches the safety valves lift setpoint.

D. Reactor steam dome pressure 800 psig, 25% rated core flow, and MCPR is 1.10 during single loop operation.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – the RPV water level safety limit is reactor vessel water level shall be greater than 15 inches above the top of active irradiated fuel.

B. Incorrect - Fuel Cladding Integrity – With the core flow < 10% rated core flow: THERMAL POWER shall be ≤ 21.7% RTP.

C. Incorrect - Fuel Cladding Integrity – The safety valves lift at 1240 psig vs. the safety limit of 1335 psig.

D. Correct – The MCPR safety limit is that with the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow: MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

Technical Reference(s): Tech Spec Sect. 2.1.1 and 2.1.2 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.11	
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases.

Proposed Question: RO Question # 70

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

- Annunciator 1C03A (C-2) POST TREAT RM 4101A/B HI RAD has alarmed
- MODE SELECT Handswitch HS-4103 is verified to be in the AUTO Position

Based on these conditions, which of the following is the expected position of the Offgas Treatment valves?

1. CV4134A, OG PREFILTERS 1F-213A/B OUTLET ISOLATION
2. CV4134B, OFFGAS CHARCOAL ADSORBER TRAIN BYPASS

A. 1. Open
   2. Open

B. 1. Open
   2. Closed

C. 1. Closed
   2. Open

D. 1. Closed
   2. Closed

Proposed Answer: B

Explanation (Optional):

A. Incorrect – CV-4134B closes to the place the charcoal adsorbers in the treatment mode

B. Correct – When the gamma radiation level in the treated offgas reaches this value, (if the auto mode of the offgas system is used), the system provides for automatic switching from the bypass mode (CV-4134B closes) to the treatment mode (CV-4134A opens).

C. Incorrect - CV-4134A opens and CV-4134B closes to the place the charcoal adsorbers in the treatment mode.

D. Incorrect - CV-4134A opens and CV-4134B closes to the place the charcoal adsorbers in the treatment mode.

Technical Reference(s): 1C03A, C-2
SD 879-1, pg 22 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # 19383

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	G3	2.3.12	
	Importance Rating	3.2		

Radiation Control: Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO Question # 71

Actions of IPOI 2, Plant Startup, are being performed. The Plant Manager has directed that all requirements of IPOI 7, Special Operations, are to be adhered to.

You have been directed to vent 1P-201A, "A" Recirc Pump while the Drywell is accessible, per OI 264, Section 10.2.

Which ONE of the following is the MAXIMUM power level permitted for this Drywell entry?

Reactor power must be …

A. below range 1 on the IRMs

B. below the point of adding heat, approximately 2% power

C. stable and below approximately 7% power

D. stable and below 30% power

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Reactor power must be below Approximately 7% power. This is plausible if the candidate assumes that the reactor must be almost subcritical to enter the drywell.

B. Incorrect – Reactor power must be below Approximately 7% power. This is plausible if The candidate assumes the limit is based on drywell temperature conditions as well as reactor power and radiation levels.

C. Correct – To protect against the radiological hazards of an operating reactor a drywell entry can be made with reactor power stable and <30 on IRM Range 10, (~7%).

D. Incorrect – Reactor power must be below Approximately 7% power unless permission is granted to exceed 7% by the Plant Manager. In this question the stem states "The Plant Manager has directed that all requirements of IPOI 7, Special Operations, are to be adhered to." Therefore, the 7% limit applies. Plausible be because the actual limit is <30 on range 10 of the IRMs (~ 7% power) and that IPOI 2 raises power level up to 35%.

Technical Reference(s): IPOI 7, Att 1, pg 4
HPP 3104, pgs 5 & 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 20492(1410)

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43

Radiological safety principles and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.34	
	Importance Rating	4.2		

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 72

Following a fire in the Control Room the crew entered AOP 915, Shutdown Outside the Control Room, and immediately evacuated the Control Room without inserting a Manual Reactor SCRAM.

In accordance with AOP 915, which ONE of the following methods is used to:

(1) shutdown the reactor?
(2) determine if the SCRAM was successful?

A. (1) Turn OFF the RPS breakers supplying Power Range Neutron Monitoring-Buses A and B.
(2) At any available computer terminal, check the control rod position printout by depressing the OD3 keys.

B. (1) Turn OFF the EPA-A1, EPA-A2, EPA-B1 and EPA-B2 circuit breakers.
(2) Locally verifying all the scram valves are open using the CRD HCU Location /Scram Valve Checklist.

C. (1) Turn OFF the RPS breakers supplying Power Range Neutron Monitoring-Buses A and B.
(2) Locally verifying all the scram valves are open using the CRD HCU Location /Scram Valve Checklist.

D. (1) Turn OFF the EPA-A1, EPA-A2, EPA-B1 and EPA-B2 circuit breakers.
(2) At any available computer terminal, check the control rod position printout by depressing the OD3 keys.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Demanding an OD3 would determine control rod position after a Scram and before the Scram reset, some control rods will go to an overtravel condition. This is a position beyond the full-in "00" indication. A control rod printout will display these overtravel positions as "-99". This display cannot be distinguished from a control rod in any other mid-stroke position where the position reed switch is not picked up. Plausible because this is an accepted way to determine control rod position.

B. Incorrect - Tripping the EPA breakers will remove power from the RPS buses and cause unnecessary isolations and is NOT used in AOP 915. Plausible because this is a quick way to scram the reactor from outside the control room.

C. Correct - IAW AOP 915; Open the door of Panel 1Y30 with a screwdriver and turn off the circuit breakers on for Power Range Neutron Monitoring System A and B. Then locally verify all scram valves open. Use Attachment 4 (CRD HCU Location /Scram Valve Checklist) for check off.

D. Incorrect - Tripping the EPA breakers will remove power from the RPS buses and cause unnecessary isolations and is NOT used in AOP 915. Plausible because this is a quick way to scram the reactor from outside the control room.
Demanding an OD3 would determine control rod position after a Scram and before the Scram reset, some control rods will go to an overtravel condition. This is a position beyond the full-in "00" indication. A control rod printout will display these overtravel positions as "-99". This display cannot be distinguished from a control rod in any other mid-stroke position where the position reed switch is not picked up. Plausible because this is an accepted way to determine control rod position.

Technical Reference(s): AOP 915, pg 23 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.13	
	Importance Rating	4.0		

Emergency Procedures / Plan: Knowledge of crew roles and responsibilities during EOP usage.

Proposed Question: RO Question # 73

The plant is operating at 100% when the following occurs:

- Drywell pressure reaches 1.25 psig and slowly rising
- Reactor water level is +165" and slowly lowering
- No automatic scram has occurred

Per OP-AA-103-1000, Reactivity Management, how should the RO respond?

A. Notify the CRS of the condition, request direction and await direction.

B. Notify the CRS of the condition, recommend a manual scram, and await direction.

C. Perform a fast power reduction using Recirc and/or control rods and announce the action to the CRS.

D. Time permitting, obtain a peer check, then initiate a manual scram and announce the action to the CRS.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – The CRO has the responsibility for obtaining a peer check and initiating a reactor scram. Plausible because there is no immediate challenge to core cooling.

B. Incorrect - Because there is no immediate challenge to core cooling the CRO has the time to obtain a peer check. Plausible because in an emergency a peer check is NOT required.

C. Incorrect - The CRO has the responsibility for obtaining a peer check and initiating a reactor scram. Plausible because there is no immediate challenge to core cooling.

D. Correct – IAW OP-AA-103-1000, Each licensed operator shall be responsible for reducing power or initiating a manual reactor scram if a key reactor safety parameter deviates from an expected condition or if it is believed necessary to assure nuclear safety. Due to time considerations, peer checking is desired but not required in these conditions.

Technical Reference(s): OP-AA-103-1000, pg 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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.14	
	Importance Rating	3.4	

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: RO Question # 74

Which ONE of the following actions requires the Control Room to notify Health Physics that areas in both Reactor Building and Radwaste may require upgrading their radiation area postings?

A. Placing a RWCU Filter Demineralizer in service

B. Operation of the Traversing In-Core Probe system

C. Swapping Fuel Pool Cooling Filter Demineralizer “A” to “B”

D. Flushing the Residual Heat Removal System for Shutdown Cooling startup

Proposed Answer: D

Explanation (Optional):

A. Incorrect – It’s plausible that backwashing a Reactor Water Cleanup filter demineralizer may affect radiation levels, however there are no precautions about changes in radiation levels during this procedure.

B. Incorrect – Plausible because operation of the Traversing In-Core Probe system may affect radiation levels in the Reactor Building. However they will NOT affect radiation levels in Radwaste

C. Incorrect – Plausible because backwashing a Fuel Pool Cooling Filter/Demineralizer requires the operator to notify HP prior and after backwashing. However there are no requirements for reposting of Rad areas when conducting this evolution.

D. Correct – The Control Room must notify Health Physics that the A[B] Loop of RHR will be warmed by flushing Reactor Water to Radwaste, followed by the startup of the A[B] Loop of RHR in the Shutdown Cooling Mode. The affected areas may require upgrading their radiation area postings.

Technical Reference(s): OI 149, pg 40 (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: N/A

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

10 CFR Part 55 Content:	55.41	12
	55.43	

Radiological safety principles and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.6	
	Importance Rating	3.0	

Equipment Control: Knowledge of the process for making changes to procedures.

Proposed Question: RO Question # 75

A plant startup is in progress following a refueling outage. A Control Room Operator is performing the initial startup of the Reactor Building Closed Cooling Water (RBCCW) system using OI 414, Reactor Building Closed Cooling Water. The operator notices the step to start the "A" RBCCW pump is incorrect as shown below:

Description	Handswitch
A TBCCW PUMP 1P-81A	HS-4829
B RBCCW PUMP 1P-81B	HS-4833
C RBCCW PUMP 1P-81C	HS-4837

Which ONE of the following statements is correct regarding the error in the procedure?

A. The RO must exit the procedure, place all equipment in a safe condition, and then discuss the error with the OSM/CRS. Until the procedure is revised, the "A" RBCCW pump may NOT be started.

B. The RO may make a field change to the procedure and start the "A" RBCCW pump. When the procedure is completed, the RO must notify the OSM/CRS of the field change for further corrective action.

C. The startup of the RBCCW system must be stopped and the error brought to the attention of the OSM/CRS. The OSM/CRS authorizes an editorial change to the procedure to permit starting the "A" RBCCW pump.

D. The startup of the RBCCW system may continue the RO must obtain concurrence from another licensed operator prior to completing the step. The editorial change must be discussed with the OSM/CRS before the end of the shift.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Typical editorial correction(s) are processed administratively and typically 'in the field'. As such, they do not require formal reviews (i.e. no PCR or Temp Change and none of the typical forms or screenings required of a permanent change. Therefore there is NO need to wait to start the pump. Plausible because most procedure changes do require submitting a procedure change request or CR which would could stop completing the procedure as written.

B. Incorrect – The work must be stopped (in this case starting the "A" RBCCW pump) and the error brought to the attention of the OSM/CRS. The OSM/CRS must authorize starting the "A" RBCCW pump. Plausible because the RO would proceed with the procedure AFTER making a field change authorized by the CRS/OSM.

C. Correct – The work must be stopped (in this case starting the "A" RBCCW pump). IAW AD-AA-100-1006, For obvious minor typographical errors, minor discrepancies in nomenclature or discrepancies involving equivalence (i.e. handswitch in STOP versus OFF, START versus HAND or MAN, etc), the OSM/CRS may, after review, allow the procedure to continue. Since the pump identification number (1P-81A) and switch identification number (HS-4829) are correct this is a typographical error. Based on this the procedure can be performed as written. Typical editorial correction(s) are processed administratively and typically 'in the field'. As such, they do not require formal reviews (i.e. no PCR or Temp Change and none of the typical forms or screenings required of a permanent change.

D. Incorrect – The work must be stopped (in this case starting the "A" RBCCW pump) and the error brought to the attention of the OSM/CRS. The OSM/CRS must authorize starting the "A" RBCCW pump, NOT another licensed operator which may be an RO. Plausible because the RO would proceed with the procedure AFTER making a field change authorized by the CRS/OSM.

Technical Reference(s): ACP 106.1, Att 5
AD-AA-100-1006, pgs 29 & 30 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295021	AA2.01
	Importance Rating		3.6

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water heatup/cooldown rate

Proposed Question: SRO Question # 76

The plant was shutdown ten (10) days ago following an extended high power run to replace a core spray pump. Plant conditions are as follows:

- “A” loop of RHR is aligned for shutdown cooling (SDC)
- The Recirculation pumps are shutdown and unavailable
- Average reactor coolant temperature is160°F
- Reactor water level is 190 inches
- The Drywell has NOT been entered
- Secondary containment is NOT established

Then,

- SDC was isolated following an RHR system leak and cannot be re-established
- RPV Level is restored and stabilized at 235 inches utilizing Core Spray “B”

Assuming that shutdown cooling CANNOT be restored:

(1) When does the plant enter Mode 3?

AND

(2) After entering Mode 3 what is the maximum amount of time before the plant is required to be in Mode 4?

A. (1) 2.2 hours
(2) 24 hours

B. (1) 2.2 hours
(2) 40 hours

C. (1) 11.5 hours
(2) 24 hours

D. (1) 11.5 hours
(2) 40 hours

Proposed Answer: A

Explanation (Optional):

A. Correct: using appendix 1 of AOP 149, the heatup rate is 25 degrees per hour. With this heatup rate, the coolant will reach 212 degrees in 2.2 hours. At 212°F Mode 3 is entered. This results in a failure to meet LCO 3.4.7. Condition A required action A.3 Be in Mode 4 within 24 hours. Condition B is also applicable but not information that is requested in the question.

B. Incorrect: 40 hours is not correct for placing the plant in Mode 4. This is plausible if the candidate incorrectly applies the Secondary Containment LCO to determine the Mode 4 time requirement.

C. Incorrect: Plausible if the candidate uses Appendix 1 of AOP 149 (vice Appendix 2) a heatup rate of 4.5 degrees per hour is obtained. In this case, Mode 3 is entered in 11.5 hours.

D. Incorrect: Plausible if the candidate uses Appendix 1 of AOP 149 (vice Appendix 2) a heatup rate of 4.5 degrees per hour is obtained. In this case, Mode 3 is entered in 11.5 hours. In addition, 40 hours is not correct for placing the plant in Mode 4. This is plausible if the candidate incorrectly applies the Secondary Containment LCO to determine the Mode 4 time requirement.

Technical Reference(s): T.S 3.4.7
T.S 3.4.8
AOP 149, App 2
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S 3.4.7
T.S 3.4.8
T.S. 3.6.4.1
AOP 149, Appendix 1 & 2

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: NA

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 #	295028	EA2.02
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor pressure

Proposed Question: SRO Question # 77

An Emergency Depressurization is in progress following high drywell air temperature conditions. Current plant conditions are as follows:

- All Control Rods are fully inserted
- Four SRVs were originally opened, but two have just failed closed
- No other SRVs can be opened
- Reactor pressure is 75 psig, lowering slowly
- Torus pressure is 10 psig and lowering slowly
- Drywell air temperature is 325°F, rising slowly
- Core Spray is injecting at rated flow
- RHR injection is NOT available
- RPV level indicator response during the depressurization was as follows:
    - Floodup Range indication pegged low but has been rising steadily and is now +190 inches and steady
    - Wide Range Yarway indication lowered steadily to +15 inches but then began rising steadily and is now +22 inches and steady
    - Fuel Zone indication lowered steadily to -40 inches but then began oscillating
    - Fuel Zone indication is now +10 inches and continuing to rise slowly.
    - Water level oscillations have stopped.

Using the information provided:

(1) What is RPV Water Level?

AND

(2) What action is required for RPV pressure control?

A. (1) +10 inches
(2) Continue depressurizing with the 2 SRVs ONLY until the plant is in cold shutdown

B. (1) +22 inches
(2) Alternate Depressurization Systems of Table 8 of ED - Emergency Depressurization must be aligned to complete the depressurization.

C. (1) -13 inches
(2) Continue depressurizing with the 2 SRVs ONLY until the plant is in cold shutdown

D. (1) -13 inches
(2) Alternate Depressurization Systems of Table 8 of ED - Emergency Depressurization must be aligned to complete the depressurization.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: RPV level is -13 inches. RPV parameters are above the Saturation Curve of graph 1. Additionally indication was received of boiling as the vessel depressurized. Per Caution 1, if boiling is suspected, 23 inches is to be subtracted from the fuel zone indication. Plausible if the candidate discounts the fuel zone indication due to the oscillations and relies on the Wide Range indication or the candidate believes that the Fuel Zone level of +10 inches is accurate because the level oscillations have stopped. If 2 of the 4 SRVs have gone closed prior to RPV pressure being within 50 psig of the torus pressure the crew must continue the depressurization with alternate depressurization systems. Plausible if the candidate assumes that the two SRVs will continue the emergency depressurization.

B. Incorrect: RPV level is -13 inches. Plausible if the candidate discounts the fuel zone indication due to the oscillations and relies on the Wide Range indication or the candidate believes that the Fuel Zone level of +10 inches is accurate because the level oscillations have stopped. However this instrument cannot be used because indication of boiling has occurred.

C. Incorrect: RPV parameters are above the Saturation Curve of graph 1 and indications of reference leg boiling has occurred. Per Caution 1, if boiling is suspected, 23 inches is to be subtracted from the fuel zone indication. This would result in a level of -13 inches. However if 2 of the 4 SRVs have gone closed prior to RPV pressure being within 50 psig of the torus pressure the crew must continue the depressurization with alternate blowdown systems. Plausible if the candidate assumes that the two SRVs will continue the emergency depressurization.

D. Correct: RPV parameters are above the Saturation Curve of graph 1 and indications of reference leg boiling has occurred. Per Caution 1, if boiling is suspected, 23 inches is to be subtracted from the fuel zone indication. This would result in a level of -13 inches. Because 2 of the 4 SRVs have gone closed, and no other SRVs will open; prior to RPV pressure being within 50 psig of the torus pressure, the crew must continue the depressurization with alternate depressurization systems.

Technical Reference(s): EOP 1, Caution 1
EOP ED, Continuous Recheck Statement above step ED-13.
(Attach if not previously provided)

Proposed References to be provided to applicants during examination:	EOP Caution 1 and associated RPV Saturation Temperature Curve EOP-ED

Learning Objective:	LP 95.00 - EOP Introduction, objective 95.00.00.14	(As available)

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

Question History:	Last NRC Exam:	NA

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 #	295016	AA2.03
	Importance Rating		4.4

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

Proposed Question: SRO Question # 78

AOP 915, Shutdown Outside the Control Room, has been entered. The Reactor was scrammed and the following conditions exist:

- RPV water level has been stabilized at +190 inches using the Feed and Condensate system
- The outboard MSIVs all indicate closed on Remote Shutdown Panel 1C-388
- RPV pressure is 900 psig and slowly rising
- Personnel are at their assigned locations and communications established

Given the above, as the Control Room Supervisor:

1) What actions should be directed to perform a plant cooldown?

   AND

2) Which of the following is correct regarding plant cooldown rate?

A. (1) Manually operate SRVs and determine cooldown rate using AOP 915, Attachments 5 and 6.
   (2) The cooldown rate should be maintained <80°F per hour.

B. (1) Manually operate SRVs and determine cooldown rate using AOP 915, Attachments 5 and 6.
   (2) Maintaining a cooldown rate is NOT required.

C. (1) Establish RCIC in pressure control mode per OI 150; Section "10.0 RCIC Operation from Outside the Control Room" and determine cooldown rate using AOP 915, Section 2, RPV Pressure/Level Control.
   (2) The cooldown rate should be maintained <80°F per hour.

D. (1) Establish RCIC in pressure control mode per OI 150; Section "10.0 RCIC Operation from Outside the Control Room" and determine cooldown rate using AOP 915, Section 2, RPV Pressure/Level Control.
   (2) Maintaining a cooldown rate is NOT required.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW AOP 915, if reactor water level can be controlled above 119.5 inches reactor pressure is maintained and the cooldown rate controlled using SRVs. A NOTE that appears in AOP 015 states "While it is desirable to limit cooldown rate to <80° F/Hr, cooldown rate cannot be controlled during Emergency Depressurization or HPCI/RCIC injection. The Technical Specification cooldown rate limit of 100° F in any 1 hr. period applies to normal plant cooldown only." Because there is NO reason stated for Emergency Depressurization or HPCI/RCIC injection the desirable cooldown rate is <80° F/Hr.

B. Incorrect - A NOTE that appears in AOP 015 states "While it is desirable to limit cooldown rate to <80° F/Hr, cooldown rate cannot be controlled during Emergency Depressurization or HPCI/RCIC injection. The Technical Specification cooldown rate limit of 100° F in any 1 hr. period applies to normal plant cooldown only." Because there is NO reason stated for Emergency Depressurization or HPCI/RCIC injection the desirable cooldown rate is <80° F/Hr.

C. Incorrect - Plausible because RCIC is cited in AOP 915, Shutdown Outside the Control Room. However, RCIC is only used for RPV water makeup and its specific operation is directed in AOP 915, NOT OI 150.

D. Incorrect - Plausible because RCIC is cited in AOP 915, Shutdown Outside the Control Room. However, RCIC is only used for RPV water makeup and its specific operation is directed in AOP 915, NOT OI 150.

Technical Reference(s): AOP 915, Section 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: NA

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	2.2.12
	Importance Rating		4.1

Knowledge of surveillance procedures. Reactor Low Water Level

Proposed Question: SRO Question # 79

The plant is operating at 100% power.

The “B” Core Spray pump was declared inoperable earlier in the shift.

Functional Testing of the HPCI LO-LO instrumentation is also scheduled for the shift as required by SR 3.3.5.1.3.

If the functional testing of the HPCI instrumentation commences as scheduled which ONE of the following is correct?

In addition to declaring the instrument channel being tested inoperable, …

A. within one hour RCIC must be verified to be operable.

B. HPCI must be declared inoperable and a 72 hour LCO must be entered.

C. if the functional test of each channel is completed within 6 hours, no additional action is required.

D. HPCI must be declared inoperable within one hour and the channel tripped if the testing is not completed within 24 hours.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Plausible if the candidate believes that HPCI must be declared inoperable to perform the test.

B. Incorrect: Plausible if the candidate believes that HPCI must be declared inoperable to perform the test. If so then LCO 3.5.1, condition H would be entered.

C. Correct: Per note 2 on page 3-3-39 (TS Section 3.3.5.1, Surveillance Requirements): When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f; and (b) for up to 6 hours for Functions other than 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f provided the associated Function (or the redundant Function for Functions 4 and 5) maintains ECCS initiation or loop selection capability.

D. Incorrect: Plausible if the candidate believes that HPCI must be declared inoperable to perform the test. As discussed above up to 6 hours is allowed to test each channel. Plausible if the candidate does not apply the note and applies the actions for each channel being inop and adds the total time allowed for the 4 instrument channels being tested (4 x 6 = 24).

Technical Reference(s): TS section 3.3.5.1, pg.39
TS section 3.5.1
STP 3.3.5.1-01, pg.3
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Spec Table 3.3.5.1-1 pages 1 through 3 only (White out all of the Allowable Values)
TS section 3.3.5.1, NO BASES
TS section 3.5.1, NO BASES

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		1
	Group #		1
	K/A #	295037	2.4.6
	Importance Rating		4.7

Conduct of Operations: 2.4.6, Knowledge of EOP mitigation strategies. (SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown)

Proposed Question: SRO Question # 80

An ATWS is in progress. The following conditions exist:

- Bypass valves failed to open and cannot be opened
- Reactor pressure is being maintained 800-1000 psig and 2 SRVs are open
- Torus water temperature is 120°F
- Torus water level is 12.8 feet
- Control rod insertion has NOT been established
- SBLC failed to inject and cannot be started
- No alternate Boron injection system is injecting

When RPV level reaches -10 inches, direction is given to re-establish injection and maintain indicated level -25 to +15 inches.

With RPV water level at -15 inches, RPV injection is re-established. Twenty (20) seconds later RPV water level is +17 inches and reactor power is 6%.

Which ONE of the following is the correct action in response to this transient?

A. Terminate and prevent injection again.

B. Perform a RPV Blowdown per EOP-ED.

C. Direct a new level control band of -10 to +15 inches.

D. Reduce the injection rate until reactor power falls below 5%.

Proposed Answer: A

Explanation (Optional):

A. Correct - Level rise outside the assigned level band (above +15) will cause reactor power to increase and exceed 5%. Override conditions are met (Power above 5%, Level above +15 inches, Torus water temp above 110°F, and an SRV is open) this requires returning to terminate and prevent injection until reactor power lowers below 5% or level is at +15 inches or the SRVs remain closed.

B. Incorrect - There is initially a 28°F margin to HCTL (120°F is the lower line on the HCTL graph), and rise in suppression pool temperature will not require RPV Blowdown at this time. Torus water level is 1 foot below the level that would require RPV Blowdown.

C. Incorrect - Level rise has caused reactor power to increase and exceed 5%. This new level band could permit power to remain at 6%. Override conditions are met to terminate and prevent injection until reactor power lowers below 5% or level is at +15 inches or the SRVs remain closed.

D. Incorrect - Override conditions are met to terminate and prevent injection again

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

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	2009 Nine Mile Point	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2009 Nine Mile Point

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 #	295003 2.4.4	
	Importance Rating		4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (Partial or Complete Loss of AC)

Proposed Question: SRO Question # 81

The plant is operating at 78% power with the Turbine Generator on line when DAEC switchyard voltage slowly lowers from 100% to 90% of normal voltages over a 5 minute time period.

Which of the following procedures are required to be entered as voltage lowers?

A. Enter AOP 304, Grid Instability; NO other procedures are required during this event.

B. Enter AOP 301, Loss of Essential Electrical Power; NO other procedures are required during this event.

C. Enter AOP 304, Grid Instability, AOP 301, Loss of Essential Electrical Power, IPOI 5 Reactor Scram, and EOP 1 – RPV Control.

D. Enter AOP 304, Grid Instability, AOP 301, Loss of Essential Electrical Power, and IPOI 5, Reactor Scram. There is NO need to enter EOPs at this time.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Plausible because AOP 304 is entered for these conditions, however with voltage this low the vital buses will trip resulting in a reactor scram as the RPS buses are de-energized.

B. Incorrect – Entry conditions are met for AOP 304, which is only exited when the essential buses 1A3 and 1A4 trip when the voltages lowers to 91.3%. When the essential buses trip it is an entry condition for AOP 301. Plausible if the candidate fails to recognize the degraded grid conditions and focuses on the loss of essential Bus power.

C. Correct – Entry conditions are met for AOP 304, which is only exited when the essential buses 1A3 and 1A 4 trip when the voltages lowers to 91.3%. When the essential buses trip it is an entry condition for AOP 301. A reactor scram will occur because of a loss of power to the RPS buses when the vital buses de-energize so the crew must enter IPOI 5. The resultant reactor SCRAM will cause RPV level to lower below the entry requirements for EOP-1 necessitating its entry.

D. Incorrect – AOP 304 is entered on the low voltage condition but is exited when the loss of the essential buses occurs and the crew must enter AOP 301. The reactor scrams because of a loss of power to the RPS buses when the vital buses de-energize so the crew must enter IPOI 5. The non-essential buses will remain energized and the essential buses are powered from the SBDGs so no loss of feedwater occurs and there is no need to enter an EOP at this time. Plausible if the candidate determines that reactor water level will be maintained through the use of Condensate and Feedwater since the non-essential busses remain energized throughout this event. The candidate may incorrectly determine that EOP-1 has no entry conditions on a loss of electrical power. However, the SCRAM from 78% power would result in exceeding the Low-Level trip setpoint requiring entry to EOP-1.

Technical Reference(s): AOP 301 pg. 18-19, AOP 304, pg.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #	295006	2.4.35
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects. SCRAM

SRO Level Justification: IPOI 5, Reactor Scram states that If all control rods are not fully inserted, EOP Support Rod Insertion Procedures may be used as authorized by the CRS. The SRO must know the operational effects of the local actions associated with each procedure in order to select the correct procedure.

Proposed Question: SRO Question # 82

The plant was at 10% power starting up, when a manual scram was inserted following an equipment failure. Conditions are as follows:

- Eight control rods failed to insert
- The Blue Scram lights are on for ALL control rods
- All scram signals have been cleared, but the scram CANNOT be reset
- The RO reports that there is insufficient cooling water pressure to drift the rods after fully opening the CRD Flow Control Valve

Which ONE of the following procedures should you direct be performed?

A. RIP 103.1, Individual Scram Test Switches.

B. RIP 103.3, Manually Drive Control Rods.

C. RIP 101.3, Vent Scram Air Header.

D. RIP 101.2, RPS Fuse Removal.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The scram signal has already caused the scram inlets and outlets for the 8 rods to open as evidenced by the blue scram lights being illuminated. Plausible if the candidate does not understand what will cause the blue scram light to turn on.

B. Correct: RIP 103.3 directs that if the scram cannot be reset and there are difficulties inserting the rods with CRD hydraulic pressure then an AO must be sent into the plant to close the CRD Charging Water Isolation Valve. This will shut off flow to the accumulators resulting in an increase in CRD drive pressure.

C. Incorrect: The air header is already vented. The scram inlets and outlets for the 8 rods have repositioned as evidenced by the blue scram lights being illuminated. Plausible if the candidate does not understand what will cause the blue scram light to turn on.

D. Incorrect: Pulling the fuses will de-energize the scram trip logic. This would be an appropriate action for an electrical ATWS. This action would prevent the RPS from resetting.

Technical Reference(s): RIP 103.3, Manually Drive Control Rods (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP SEG 6, objective, 6.07.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		2
	K/A #	295029	EA2.01
	Importance Rating		3.9

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

Proposed Question: SRO Question # 83

Following a loss of coolant accident the following conditions exist:

- RPV level is +170 inches, rising slowly
- Condensate is injecting and maintaining level
- RPV pressure is 200 psig, lowering slowly
- RHR "A" and "B" are aligned for both Torus and Drywell sprays
- Core Spray is in standby
- Drywell pressure is 16 psig, lowering slowly
- Drywell temperature is 240°F lowering slowly
- Torus pressure is 15 psig lowering slowly
- Torus water temperature is 120°F, rising slowly
- Torus water level is 13.5 feet rising slowly

IAW EOP 2, Primary Containment Control, which of the following actions is required?

A. Secure Drywell sprays and emergency depressurize

B. Emergency depressurize while continuing Drywell sprays

C. Secure Drywell sprays and terminate injection from condensate and inject with low pressure ECCS

D. Terminate injection from condensate and inject with low pressure ECCS while continuing Drywell sprays

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Emergency Depressurization is not required at this time. Plausible in that Torus parameters are close to exceeding the PSP curve. The required action to secure injection from condensate and shift injection to the low pressure ECCS will terminate the torus level rise and prevent exceeding the PSP curve.

B. Incorrect: Emergency Depressurization is not required at this time.

C. Correct: When Torus level cannot be maintained below 13.5 feet, step T/L-11 requires that the drywell sprays be secured. This action is required because the torus to drywell vacuum breakers are covered and continued spraying could result in exceeding the negative design pressure rating of the containment. Additionally, step T/L 12 requires that injection from sources external to the primary containment be terminated if not required for core cooling. Since RPV pressure is within the capacity of low pressure ECCS, RPV level control should be shifted to LP ECCS to stop the torus level rise.

D. Incorrect: Step T/L-11 requires that the drywell sprays be secured.

Technical Reference(s): EOP 2 step T/L-11 and 12
Print in COLOR (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP Graph 5

Learning Objective: SEG 108, LO 95.59 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		2
	K/A #	295009	2.4.45
	Importance Rating		4.3

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm: Low Reactor Water Level

Proposed Question: SRO Question # 84

During a major loss of coolant accident the following conditions exist:

- All control rods are inserted
- RPV level is steady at -30”
- Core Spray pump “B” is injecting at 3100 gpm
- RHR pumps “A” and “B” are aligned for Torus and Drywell sprays
- Torus pressure is 31 psig and slowly lowering
- ALL other injection sources are unavailable

Then, alarm 1C03C (A-2) “B” CORE SPRAY PUMP 1P-211B TRIP OR MOTOR OVERLOAD, annunciates. Operators report that Core Spray Pump “B” amps pegged immediately prior to the pump trip.

Which ONE of the following is required by the EOPs?

A. SAG entry is required. Coordinate with the TSC to exit BOTH EOP 1 AND EOP 2 and enter SAGs.

B. SAG entry is required. Coordinate with the TSC to exit EOP 1 and enter SAGs Continue Torus and Drywell Sprays in accordance with EOP 2.

C. Secure all containment sprays and direct all RHR flow to the RPV to restore RPV level greater than -25 inches. Make preparations to vent the containment in accordance with EOP 2.

D. Continue Torus and Drywell Sprays in accordance with EOP 2. If RPV level lowers to -39 inches, secure all containment sprays and direct all RHR flow to the RPV to restore RPV level greater than -39 inches. Make preparations to vent the containment in accordance with EOP 2.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Once the Core Spray pump tripped, adequate core cooling via spray cooling was lost. EOP 1 step RC/L-12 directs that injection be maximized with all available injection systems in an attempt to restore level to above -25 inches. Only after this attempt has been proven unsuccessful is SAG entry required. Additionally, once SAG entry is required, all EOPs are exited.

B. Incorrect: Once the Core Spray pump tripped, adequate core cooling via spray cooling was lost. EOP 1 step RC/L-12 directs that injection be maximized with all available injection systems in an attempt to restore level to above -25 inches. Only after this attempt has proven unsuccessful is SAG entry required.

C. Correct: Once the Core Spray pump tripped, adequate core cooling via spray cooling was lost. EOP 1 step RC/L-12 directs that injection be maximized with all available injection systems in an attempt to restore level to above -25 inches. Additionally, EOP 2 directs that only those pumps not required for core cooling be utilized for containment sprays. Since torus pressure was 31 psig, it will begin to rise once sprays are secured. The only available action then is to vent the containment to maintain pressure less than the limit of 53 psig.

D. Incorrect: Once the Core Spray pump tripped, adequate core cooling via spray cooling was lost. EOP 1 step RC/L-12 directs that injection be maximized with all available injection systems in an attempt to restore level to above -25 inches.

Technical Reference(s): EOP 1 step RC/L-12 (Attach if not previously provided)

EOP 2 steps PC/P-9 and 10

Proposed References to be provided to applicants during examination: None

Learning Objective: LP SEG 98, objective 6.74.08 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		2
	K/A #	295012	AA2.01
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature

Proposed Question: SRO Question # 85

A LOCA is in progress with the following conditions:

- RPV water level is being maintained at +180 inches with Condensate and Feedwater
- Both recirc pumps are tripped
- Containment sprays are currently unavailable
- Drywell temperature reaches 280°F and continues to rise slowly.

Ten (10) minutes later Drywell temperature and pressure are now reported to be 290°F and 12 psig, both slowly rising and within the bounds of the Drywell Spray Initiation Limit.

Which ONE of the following describes the actions that are to be taken to control primary containment temperature if Drywell sprays become available per EOP 2?

A. Enter ED-Emergency Depressurization; after the reactor is depressurized Drywell sprays are NOT required.

B. Enter ED-Emergency Depressurization, spray the Drywell to lower Drywell temperature. If Drywell temperature lowers below 280°F, exit ED-Emergency Depressurization and re-enter EOP-1 pressure control.

C. IAW EOP 2, Primary Containment Control, spray the Drywell to lower Drywell temperature and pressure. If Drywell temperature and pressure do NOT lower, exit EOP 1 Pressure Control and enter ED-Emergency Depressurization

D. IAW EOP 2, Primary Containment Control, spray the Drywell to lower Drywell temperature and pressure. Exit EOP 1 Pressure Control and enter ED-Emergency Depressurization regardless of Drywell temperature response.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: An immediate Emergency Depressurization is not required. Plausible in that temperature is above the value associated with the Emergency Depressurization step. Additionally the decision not to spray the drywell is also plausible if the candidate does not understand the meaning of a “Before” step. The direction to spray the drywell is “Before” the drywell temperature reaches 280°F.

B. Incorrect: An immediate Emergency Depressurization is not required. Plausible in that temperature is above the value associated with the RPV ED step.

C. Correct: Drywell and temperature and pressure are within the DSIL curve. Although drywell temperature is above the value requiring a blow down, the associated step directs that when temperature “cannot be restored and maintained” below 280°F, only then is a blow down performed. As discussed in the EOP bases, the step allows the operator to attempt to restore the drywell temperature below 280°F. Since not all the steps have been performed to prevent the blow down the action of initiating drywell sprays should first be attempted. If after attempting drywell sprays, drywell temperature cannot be restored below 280°F, then Emergency Depressurization is required.

D. Incorrect: Emergency Depressurization is not immediately required after spraying the drywell. Plausible in that temperature is above the value associated with the Emergency Depressurization step.

Technical Reference(s): EOP 2 step DWT-6 (Attach if not previously provided)

EOP 2 Bases page 39

Proposed References to be provided to applicants during examination: None

Learning Objective: SEG 108, objective 6.63.08 (As available)

Question Source: Bank # WTS # 11851

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 NMP2

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 #	215005	A2.08
	Importance Rating		3.2

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions Faulty or erratic operation of detectors/systems

Proposed Question: SRO Question # 86

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

- APRM “A” has failed upscale and is bypassed
- APRM “D” is bypassed for normal operation IAW OI 878.4, APRM System
- Several LPRM detectors associated with APRM “E” have failed and have been bypassed

Then, LPRM 5A-32-17 fails downscale.

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

(1) Prior to any operator action, what will be the impact of the LPRM failure on APRM "E"?

AND

(2) Which of the below actions is required for this condition?

APRM E Detectors

LPRM	(Status)
2A-16-33	Bypassed
5A-32-17	Downscale
3A-16-25	OK
6A-32-09	Bypassed
3B-24-25	OK
4B-08-09	OK
4B-08-17	Bypassed
5B-40-17	OK
3B-24-33	OK
3C-32-33	Bypassed
4C-16-17	OK
1C-16-41	OK
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. (1) APRM “E” output will lower.
(2) Enter OI 878.3, LPRM System, bypass LPRM 5A-32-17, verify or adjust AGAFs as required and continue plant operation without any additional restrictions.

B. (1) APRM “E” output will remain the same.
(2) Enter OI 878.3, LPRM System, bypass LPRM 5A-32-17, verify or adjust AGAFs as required and continue plant operation without any additional restrictions.

C. (1) APRM “E” output will lower.
(2) Enter Tech Spec LCO 3.3.1.1, RPS Instrumentation. If APRM "A" or "E" is not restored within the required completion time, insert a trip on RPS “A”.

D. (1) APRM “E” output will remain the same.
(2) Enter Tech Spec LCO 3.3.1.1, RPS Instrumentation. If APRM "A" or "E" is not restored within the required completion time, insert a trip on RPS “A”.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: APRM “E” is inoperable due to having only one operable LPRM at the “A” level. IAW OI 878.4 to be considered operable, APRM E must have at least 13 LPRM inputs. Each APRM must have at least 2 LPRM inputs per level. Plausible in that the APRM does satisfy the requirement for the overall number of LPRMs in that 13 of the 20 assigned LPRMs is operable. If the candidate believes that the APRM is operable then the required action would be to bypass the LPRM and adjust gains as required.

B. Incorrect: The APRM output would lower. Plausible in that, unlike the Rod Block Monitor, there are no input trip units that monitor the status of the LPRMs inputting into the APRM. Therefore the output of the downscale LPRM would be included into the overall average causing the APRM output to lower. Additionally the APRM would be inoperable as described in explanation “A”.

C. Correct: The output of the APRM would lower as described in explanation “B”. IAW OI 878.4 to be considered operable, APRM E must have at least 13 LPRM inputs. Each APRM must have at least 2 LPRM inputs per level. Therefore APRM “E” is inoperable. With APRM “A” also inoperable only one APRM is operable for RPS “A”. TS LCO3.3.1.1, RPS Instrumentation and associated table requires a minimum of 2. Therefore Condition “A” of the LCO is not satisfied and the required action is to place the channel or trip system in trip within 12 hours.

D. Incorrect: The output of the APRM would lower as described in explanation “B”.

Technical Reference(s): TS LCO 3.3.1.1 and associated TS Table. (Attach if not previously provided)
APRM System Description pages 17 through 19
OI 878.4, P&L # 6

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI Bank 12910
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2011 Pilgrim

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 #	223002	A2.03
	Importance Rating		3.0

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

K/A Justification: Although not an obvious two part question, part (a) of the K/A is tested in that the SRO must predict the impact of the trip settings on the PCIS and then determine what action is required per Tech Specs to mitigate the effects as required by part (b) of the K/A.

Proposed Question: SRO Question # 87

The plant is at 100% power. I & C then reports that the A1 Trip Channel for the Group 1 and Group 2 PCIS are currently set to trip at:

- Group 1 A1 (Reactor Vessel Water Level Low Low Low) Trip Channel: 35 inches
- Group 2 A1 (Reactor Vessel Water Level Low) Trip Channel: 160 inches

Assuming all other trip channels are set to trip at their nominal values, what is the maximum time allowed before the A1 Trip Channels must be tripped in order to restore operability?

A. Group 1: 12 hours
Group 2: 12 hours

B. Group 1: 12 hours
Group 2: 24 hours

C. Group 1: 24 hours
Group 2: 12 hours

D. Group 1: 24 hours
Group 2: 24 hours

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Group 1 Channel A1 isolation logic initiates MSIV closure on Low-Low-Low level. Per TS Table 3.3.6.1-1, this function is identified as Function 1.a. and must be set to trip ≥ 38.3 inches. IAW LCO 3.3.61 if one or more required channels are inoperable, then that channel must be placed in trip within 24 hours if it is associated with any function other than 2.a, 2.b, 6.b, and 6.c.

B. Incorrect: The completion time for tripping the Group 1 Channel A1 trip channel is 24 hours as described above. Additionally, Group 2 Channel A1 isolation logic initiates a Primary Containment Isolation on Low level. Per TS Table 3.3.6.1-1, this function is identified as Function 2.a. and must be set to trip ≥ 165.6 inches. IAW LCO 3.3.61 if one or more required channels are inoperable, then that channel must be placed in trip within 12 hours if it is associated with function 2.a.

C. Correct: The completion time for tripping the Group 1 Channel A1 trip channel is 24 hours as described above. The completion time for tripping the Group 2 Channel A1 trip channel is 12 hours as described above.

D. Incorrect: The completion time for tripping the Group 2 Channel A1 trip channel is 12 hours as described above.

Technical Reference(s): TS LCO 3.3.6.1 and Table 3.3.6.1-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS LCO 3.3.6.1 and Table 3.3.6.1-1

Learning Objective: LP 94.18, objective 5.18.01.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		2
	Group #		1
	K/A #	239002	2.4.18
	Importance Rating		4.0

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

Proposed Question: SRO Question # 88

RPV Flooding is being executed following a complete loss of RPV level indication.

Plant conditions are as follows:

- Many control rods are still fully withdrawn
- The switches for four (4) ADS valves have been placed in OPEN
- Reactor pressure is 700 psig and lowering

Given these conditions what action is required?

Wait until RPV pressure is below the Minimum Steam Cooling Pressure (MSCP), then…

A. slowly raise injection to maintain at least one SRV open and reactor pressure above, but as close to the MSCP as possible.

B. slowly raise injection to the RPV to maintain all four SRVs open and reactor pressure below, but as close to the MSCP as possible.

C. maximize injection to the RPV to maintain all four SRVs open with reactor pressure above, but as close to the MSCP as possible.

D. maximize injection to the RPV to maintain at least one SRV open with reactor pressure below, but as close to the MSCP as possible.

Proposed Answer: A

Explanation (Optional):

A. Correct – When RPV pressure is below the Minimum Steam Cooling Pressure (MSCP) injection flow must be slowly raised. With 4 ADS valves opened, the Minimum Steam Cooling Pressure is 160 psig. With one SRV open the MSCP is 680 psig. As long as RPV pressure is maintained above the MSCP for the number of SRVs open the core will be cooled by submergence or steam cooling.

B. Incorrect - When RPV pressure drops below the Minimum Steam Cooling Pressure, steam flow may no longer be sufficient to provide adequate core cooling. All four SRVs are NOT required to be open.

C. Incorrect - If injection is maximized RPV water level, reactor power, and RPV pressure will rise. Injection must be throttled to reduce pressure and the subsequent power increase. Injection must NOT be maximized and all injection must be from ATWS preferred flooding systems. All four SRVs are NOT required to be open.

D. Incorrect - If injection is increased, RPV water level, reactor power, and RPV pressure will rise. Injection must be throttled to reduce pressure and the subsequent power increase. Injection must NOT be maximized and all injection must be from ATWS preferred flooding systems. As long as RPV pressure is maintained above MSCP the core will be cooled by submergence or steam cooling.

Technical Reference(s): RPV/F – RPV Flooding RPV/F-8
RPV/F EOP Bases pgs 16, 17, 18 (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: N/A

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 #	211000	2.2.25
	Importance Rating		4.2

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 89

During operation at 100% power an operator reports the heat tracing on the suction side of the Standby Liquid Control (SBLC) pumps is damaged and inoperable. The following conditions exist:

- SBLC Tank Concentration is 14%.
- SBLC Tank Volume 3200 gallons.
- Reactor Building Ambient Temperature at SBLC Pumps is 72°F.

Which ONE of the following describes the condition of the SBLC system per Technical Specifications?

A. OPERABLE as long as Reactor Building Temperature remains at its current value which is based on preventing Boron from precipitating out of solution inside system components.

B. INOPERABLE because the Boron Solution Concentration is below the minimum allowable. A 7 day LCO is required, which is based on injecting 780 ppm of Boron solution into the reactor core.

C. INOPERABLE because the Boron Solution Concentration is below the minimum allowable. An 8 hour LCO is required, which is based on injecting 780 ppm of Boron solution into the reactor core.

D. OPERABLE because Technical Specifications allows 24 hours to restore the heat tracing to operability as long as Reactor Building Temperature is above the minimum. Based on preventing Boron from precipitating out of solution and lowering the effective Boron concentration.

Proposed Answer: A

Explanation (Optional):

A. Correct – SR 3.1.7.3 (verify solution and piping temps >70°F) WILL be met with RB Temperature at 72°F in the vicinity of SBLC components. Heat Trace functionality is not specified as a Surveillance Requirement.

B. Incorrect – The SBLC is operable, plausible; with ONE subsystem INOPERABLE, a 7 day LCO would be applicable

C. Incorrect - The SBLC is operable, plausible; with BOTH subsystems INOPERABLE, an 8 hour LCO would be applicable

D. Incorrect – There is no Technical Specifications requirement for the heat tracing to be operable. Plausible because surveillance testing for suction piping temperatures must be taken every 24 hours. Additionally the tank heat is required to maintain SBLC tank temperature which prevents tank temperature from lowering and permitting boron to precipitate out of solution within the tank.

Technical Reference(s): T.S. Bases 3.1.7
T.S. 3.1.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S. 3.1.7 Figures 3.1.71 and 3.1.7.-2-

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2008 Nine Mile Point 2

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 #		2
	Group #		1
	K/A #	217000	A2.15
	Importance Rating		3.8

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: Steam line break

Proposed Question: SRO Question # 90

Given the following conditions:

- A RCIC steam leak has occurred in the Reactor Building
- Efforts to isolate the leak are unsuccessful
- Water level in the RCIC Room (LI 3769) is 4 inches
- RCIC Emergency Cooler outlet temp (TR/TDR 2425 Ch 1) is 301°F
- RCIC Room ambient (TR/TDR 2425 Ch 2) is 325°F

Which ONE of the following correctly completes the following statement?

Based on the conditions above, entry into Emergency Depressurization…

A. is currently required because the integrity of the secondary containment is threatened.

B. is currently required because the continued operability of safety related equipment is threatened.

C. will be required if Torus East Catwalk Ambient temperature exceeds 165°F because the continued operability of safety related equipment is threatened.

D. will be required if the water level in the "A" RHR & CS SECR rises above its Max safe limit because the integrity of the secondary containment is threatened.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Emergency Depressurization is not currently required. Plausible in that both temperatures are above their Max Safe Operating Values but they are within the same area. Emergency Depressurization is performed if Max Safe values are exceeded in 2 different areas.

B. Incorrect: Emergency Depressurization is not currently required. Plausible in that both temperatures are above their Max Safe Operating Values but they are within the same area. Emergency Depressurization is performed if Max Safe values are exceeded in 2 different areas.

C. Correct: When the Torus East Catwalk Ambient temperature exceeds 165°F, Max Safe limit are now exceeded in two different areas and Emergency Depressurization is required. Maximum Safe Limits 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.

IAW NUREG 1021, Section 401 2.a. When selecting or writing questions for K/As that test coupled knowledge or abilities (e.g., the A.2 K/A statements in Tiers 1 and 2 and a number of generic K/A statements, such as 2.4.1, in Tier 3), try to test both aspects of the K/A statement. If that is not possible without expending an inordinate amount of resources, limit the scope of the question to that aspect of the K/A statement requiring the highest cognitive level (e.g., the (b) portion of the A.2 K/A statements) or substitute another randomly selected K/A. Predicting the impact of a steam line break on the RCIC system would be an RO knowledge. Rather than complicate the question Because we felt this was a valid K/A to test on we have only tested the second part of the K/A.

D. Incorrect: Emergency Depressurization is required when the same parameter exceeds the Max Safe Values in 2 or more areas. Plausible in that a Max Safe value has been exceeded in but a different parameter.

Technical Reference(s): EOP-03
EOP-03, Bases, pg 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP 3, Table 6 (Table only)

Learning Objective: (As available)

Question Source: Bank # WTSI 12939

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2004 Nine Mile Point 1

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 #		2
	K/A #	223001	A2.11
	Importance Rating		3.8

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Abnormal suppression pool level

Proposed Question: SRO Question # 91

The plant was operating normally at power when a loss of coolant accident occurred. Conditions after the scram are as follows:

- All control rods are inserted
- Drywell pressure is 5.5 psig and rising slowly
- Drywell temperature is 250°F and rising slowly
- RPV water level is 45 inches and slowly lowering
- Torus water level is 7.1 feet and slowly lowering
- RPV pressure 700 psig and slowly lowering
- Torus water temperature is 120°F and rising slowly

Per the EOPs, which ONE of the following is required AND why?

A. Secure Core Spray pumps to prevent cavitation damage of the pumps.

B. Place RHR in Drywell Spray to lower Drywell temperature and pressure.

C. Emergency depressurize due to compromise of pressure suppression capability.

D. Cycle SRV's to maintain RPV pressure below the heat capacity limit.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Plausible because with the torus level so low high core spray flows would be limited by the vortex limits. However if the flow was affected by the vortex limits core spray flow should be lowered. With reactor water level lowering the core spray pumps should NOT be secured.

B. Incorrect – Drywell pressure is too low to allow drywell sprays in that drywell temperature and pressure are not within the bounds of the Drywell Spray Initiation Limit (DSIL). Plausible in that the EOP action to initiate drywell spray would be appropriate if within the bounds of the DSIL.

C. Correct - Per EOP bases Torus water level must be maintained above the bottom of the downcomer vent openings (7.1 feet) to ensure that steam discharged from the drywell into the torus following a primary system break will be adequately condensed. If a primary system break were to occur with torus water level below the bottom of the downcomers, pressure suppression capability would be unavailable and torus pressure could exceed the Primary Containment Pressure Limit.

D. Incorrect - Plausible if the candidate does NOT realize that HCL is substantially higher for this reactor pressure. With RPV pressure between 700 and 800 psig, HCL limit is ~150°F even when including the low torus water level.

Technical Reference(s): EOP-2
EOP-2, Bases, pg 12
Print in COLOR (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: N/A

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 #		2
	K/A #	288000 2.2.40	
	Importance Rating		4.7

Equipment Control: Ability to apply technical specifications for a system. (Plant Ventilation)

Proposed Question: SRO Question # 92

With the plant operating at 100% power the fan belts on the "B" RHR/CS corner room cooler, 1V-AC-11 must be replaced. The Technical Requirements Manual states:

TLCO 3.5.2 The following ECCS and RCIC unit coolers shall be OPERABLE:

a. One RCIC room unit cooler;

b. One HPCI room unit cooler; and

c. Two CS/RHR room unit coolers.

APPLICABILITY: When the associated pumps are required to be OPERABLE.
ACTIONS

---NOTES---

1. Separate condition entry is allowed for each unit cooler.

2. TLCO 3.0.4.b is N/A for HPCI and RCIC Room Coolers.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required unit room cooler inoperable.	A.1 Declare the associated pump(s) inoperable.	Immediately

Which ONE of the following actions is required by Technical Specifications?

A. No action is required because room cooler 1V-AC-12 remains operable.

B. 1V-AC-11 must be restored to service immediately because ONLY one RHR/CS Room Cooler remains operable.

C. 1V-AC-11 must be restored to service within 72 hours because the "B" and "D" RHR and "B" Core Spray pumps must be declared inoperable

D. Place the plant in Mode 3 within 12 hours because with one cooler inoperable the TRM requirement for two CS/RHR room unit coolers is NOT met and T.S. 3.5.1.E applies.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – There are only two RHR/CS Room Coolers, one for each ECCS division. With one cooler out of service TRM requires declaring the associated pump(s) inoperable, in this case the B and D RHR and B Core Spray pumps.

B. Incorrect – This would apply if both loops of RHR and Core Spray were affected by the loss of one cooler. It is plausible because the TRM states than two room coolers must be operable and with one inoperable only one remains operable.

C. Correct – With 1V-AC-11 inoperable the B and D RHR and B Core Spray pumps must be declared inoperable this requires entry into TS 3.5.1.C which requires restoring one of the two systems to operable within 72 hours. Since both these systems must be considered inoperable until the cooler is returned to service the cooler must be returned to service within 72 hours.

D. Incorrect – This would apply if the candidate literally reads the TRM requirement as both (Two) CS/RHR room unit coolers must be operable. This would imply that all the both RHR and CS systems are considered inoperable and therefore T.S. 3.5.1.E would apply. However as stated in the TRM Separate condition entry is allowed for each unit cooler, therefore only one of the coolers and associated RHR/CS systems are considered inoperable.

Technical Reference(s): TRM 3.5.2
TS 3.5.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: T.S. 3.5.1

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		2
	Group #		2
	K/A #	215001	2.4.41
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications. (Traversing In-core Probe)

Proposed Question: SRO Question # 93

The plant was operating at 100% power when the following events occurred:

Time	Event
0100	Traversing Incore Probe (TIP) traces are in progress.
0108	Both seals on a Reactor Recirc Pump fail.
0110	The reactor scrams when Drywell pressure exceeds 2.0 psig. RPV level lowers to +160 inches before recovering to the normal range.
0115	BOP Operator reports that the Recirc Pump cannot be isolated and that Drywell pressure is now 3.0 psig and rising slowly. Reactor Operator reports the following indications for the TIP system: • Two TIPs have withdrawn to their shields and their ball valves have closed • The third detector has NOT withdrawn and its ball valve remains open
0120	Reactor Operator reports the detector will not move and the ball valve CANNOT be manually closed.
0125	Reactor Operator reports the shear valve failed to fire.

In accordance with the EAL for the Fission Product Barrier Matrix which ONE of the following contains ALL of the required Emergency Plan classifications?

A. An UNUSUAL EVENT must be declared by 0125.

B. An ALERT must be declared by 0125.

C. An UNUSUAL EVENT must be declared by 0125.
An ALERT must be declared by 0140.

D. An ALERT must be declared by 0125.
A SITE AREA EMERGENCY must be declared by 0140.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: A SITE AREA EMERGENCY was exceeded at time 0125 when the shear valve failed to fire and needed to be declared by 0140 based on EAL FS1. Plausible if the candidate only recognizes that the RCS barrier was lost when drywell pressure exceeded 2.0 psig and determines that Unusual Event FU1 was exceeded. This would also be incorrect in that an Alert EAL was also exceeded when the RCS barrier was lost based on EAL FA1.

B. Incorrect: A SITE AREA EMERGENCY was exceeded at time 0125 when the shear valve failed to fire and needed to be declared by 0140 based on EAL FS1. Plausible if the candidate only recognizes that the RCS barrier was lost when drywell pressure exceeded 2.0 psig and determines that ALERT EAL FA1 was exceeded.

C. Incorrect: A SITE AREA EMERGENCY was exceeded at time 0125 when the shear valve failed to fire and needed to be declared by 0140 based on EAL FS1. Plausible if the candidate recognizes that the PC barrier was lost at time 0125 and thinking that the event should be upgraded to an ALERT by 0140 based on the second barrier failing.

D. Correct: The RCS barrier was lost when drywell pressure exceeded 2.0 psig. This resulted in ALERT EAL FA1 being exceeded at time 0110. The ALERT was required to be declared by 0125.

   Then when the shear valve failed to fire, the PC Barrier was lost at time 0125. This resulted in a SITE AREA EMERGENCY EAL being exceeded based on EAL FS1 due to having lost two barriers. The SITE AREA EMERGENCY was required to be declared within the next 15 minutes or time 0140.

Technical Reference(s): EAL-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EAL-01, only the portion Fission Product Barriers

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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:

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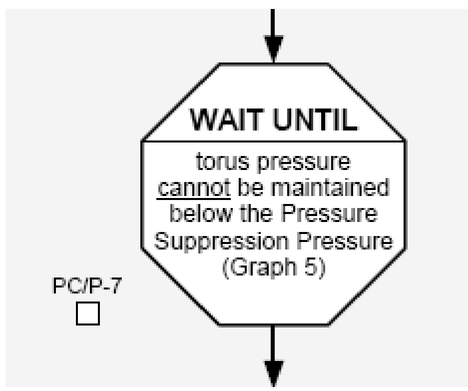
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.20
	Importance Rating		4.6

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: SRO Question # 94

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

- A spurious Group I isolation occurred which cannot be reset.
- As a result of the pressure spike, a break occurred in the Recirculation system piping.
- While executing EOP-2, Primary Containment Control, it is determined that Torus pressure is approaching but has not yet reached the Pressure Suppression Pressure (PSP) Curve.



Under these conditions, Emergency Depressurization...

A. is required when you make the determination that the PSP limits will be exceeded.

B. should be anticipated by depressurizing the RPV using Alternate Pressure Control Systems.

C. is NOT allowed until the PSP curve limits are exceeded but then must be executed regardless of any anticipated trends.

D. is NOT required if the PSP limits are exceeded provided you believe you will be able to restore parameters based on anticipated trends.

Proposed Answer: A

Explanation (Optional):

A. Correct - EOP-2, Step PC/P-7 is a Hold Point Figure regarding the PSP, this step precludes subsequent actions until the conditions listed in the Hold Point (Exceeding PSP) are met. When the conditions in the Hold Point are met, the operator continues to the next step in the flowchart and performs it. This hold point is further clarified by the Point of Emphasis, The logic phrase CAN/CANNOT BE MAINTAINED dictates that the required action should be initiated before or at the identified critical value. The value of the identified parameter(s) is/is not able to be kept above/below specified limits. This determination includes making an evaluation that considers both current and future system performance in relation to the current value and trend of the parameter(s). As specifically regards "cannot," this does not imply that the actual value of the parameter must first pass the specified limit. Proceeding actions in the EOP flow path should be directed prior to or at the stated condition.

B. Incorrect – Emergency Depressurization can only be anticipated using the turbine bypass valves. Plausible since the MSIVs are closed.

C. Incorrect - The logic phrase CAN/CANNOT BE MAINTAINED dictates that the required action should be initiated before or at the identified critical value.

D. Incorrect - There is no provision for restoring PSP, when it is believed PSP cannot be maintained. Emergency Depressurization must be executed.

Technical Reference(s): EOP-02, Step PC/P-7 Bases
Bases Flowchart Use pg 30 (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: N/A

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:

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

When proposing a design change to DAEC, the 10CFR 50.59 process determines if...

A. the change may be accomplished without prior NRC approval.

B. an evaluation for compliance with NRC Reg Guides is required.

C. a Technical Specification revision is required after implementation.

D. a Technical Specification Bases change is required after implementation.

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 - Plausible because a design change may be in response to a Reg Guide however the 50.59 review does NOT evaluate compliance with NRC Reg Guides.

C. Incorrect - Plausible because the evaluation may determine that a Technical Specifications is required. It does not, however, always require a TS revision, and it would NOT allow a TS change after the fact for a facility change. An UFSAR change is sometimes allowed to take up to 2 years to make, and this choice is additionally plausible if the candidate transferred this knowledge to TS.

D. Incorrect - Plausible because a design change may involve a Technical Specification Bases change, however 50.59 evaluates for license (TS) amendment requirements NOT Bases revisions. A Technical Specification Bases is not part of the determination.

Technical Reference(s): ACP-103.2 rev 39, pg 3,
ACP 103.0, rev 23, page 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: SRO 1.11.02.06 (As available)

Question Source: Bank # DAEC 50652

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 DAEC (# 95)

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	G3	2.3.13
	Importance Rating		3.8

Radiation Control: Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: SRO Question # 96

During movement of an irradiated fuel bundle from the reactor to the spent fuel pool, a reactor primary system piping break results in Fuel Pool level slowly lowering below 36 feet.

- The refueling platform has entered the spent fuel pool area from the reactor cavity.
- No radiation alarms have occurred at this time.

Which ONE of the following actions is required?

A. Return the bundle to its original position then suspend all movement of irradiated fuel.

B. Immediately stop fuel handling and evacuate via the safest exit to the reactor building 4th floor.

C. Immediately place the bundle in the spent fuel pool and suspend movement of irradiated fuel.

D. Immediately evacuate nonessential personnel; essential personnel place fuel into its original position and evacuate via the safest exit to the reactor building 4th floor.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The fuel bundle must be taken to the spent fuel pool.

B. Incorrect – There is NO requirement to evacuate the refuel floor at this time, Plausible because AOP 981 Fuel Handling Event directs evacuating the floor, however the AOP anticipates high radiation on the refuel floor.

C. Correct –Whenever Spent Fuel Pool Level is less than 36 feet, immediately suspend movement of irradiated fuel and place any load suspended in the Spent Fuel Pool into a safe configuration.

D. Incorrect – There is NO requirement to evacuate the refuel floor at this time. Plausible because AOP 981 Fuel Handling Event directs evacuating the floor, however the AOP anticipates high radiation on the refuel floor. Additionally the fuel bundle must be taken to the spent fuel pool.

Technical Reference(s): RFP 403, P & L 2.27, pg 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: N/A

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 #		4
	K/A #	G4	2.4.8
	Importance Rating		4.5

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.

Proposed Question: SRO Question # 97

With the plant at 100% power drywell temperature and pressure begin to rise due to a partial loss of Well Water. The following sequence occurs:

- AOP 408 Well Water System Abnormal Operation is entered
- Annunciator 1C05B (B-1), PRIMARY CONTAINMENT HI/LO PRESSURE alarms
- Drywell pressure is reported as 1.55 and rising slowly
- Drywell temperature is reported as 130 °F and rising slowly

Then …

- Drywell pressure is reported as 1.70 and rising slowly
- Drywell temperature is reported as 155 °F and rising slowly

Which ONE of the following is correct regarding additional procedures to be used to mitigate the event?

A. EOP 2 Primary Containment Control must be entered. All AOPs must be exited. Response to the event is as directed by the EOP.

B. EOP 2 Primary Containment Control must be entered. AOP 573, Primary Containment Control may also be entered as determined by the Shift Manager. If a conflict arises, the actions of the event specific AOPs take precedence.

C. EOP 2 Primary Containment Control must be entered. AOP 573, Primary Containment Control may also be entered as determined by the Shift Manager. AOP execution may continue provided that the actions do not conflict with EOP actions.

D. EOP 2 Primary Containment Control and AOP 573 Primary Containment Control must be entered. However initial response to the event is IAW AOP 408 and 573. Execution of EOP actions may be delayed until all applicable actions of the AOPs have been completed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The AOP execution can continue provided the actions do not conflict with the EOP actions and it is approved by the shift manager. Plausible in that the EOP is the higher tier document in the hierarchy of procedures.

B. Incorrect: If a conflict arises the EOP takes precedent. The Shift Manager is not authorized to deviate from an EOP action.

C. Correct: IAW ACP 1410.1, 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. If other plant 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.

D. Incorrect: There is no allowance to defer EOP actions when in an AOP.

Technical Reference(s): ACP 1410.1, pg 17
EOP 2
AOP 573
(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: NA

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.1
	Importance Rating		4.4

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: SRO Question # 98

Today is January 8, 2013. The plant has been shutdown for 5 days following a forced outage. The last plant startup was conducted on November 14, 2011.

In preparation for today's reactor startup STP 3.3.2.1-05 RWM Control Rod Sequence Verification is being performed. During the performance of the STP the RWM "locks-up".

Based on the above:

(1) What method specified in OI 878.8, RWM can be used in an attempt to reset the RWM "lock-up"?

AND

(2) If the RWM will not reset, will Tech Specs allow control rod withdrawal?

A. (1) Reset the RWM by de-energizing and then re-energizing its power supply.
(2) No, because using a 2nd licensed operator or other qualified individual is only allowed if ≥ 12 rods are withdrawn.

B. (1) Reset the RWM by de-energizing and then re-energizing its power supply.
(2) Yes, but only if rod movements are verified by a 2nd licensed operator or other qualified individual.

C. (1) Reset the RWM by placing the RWM keylock Mode Switch on panel 1C05 in TEST then back to the OPERATE position.
(2) Yes, but only if rod movements are verified by a 2nd licensed operator or other qualified individual.

D. (1) Reset the RWM by placing the RWM keylock Mode Switch on panel 1C05 in TEST then back to the OPERATE position.
(2) No, because using a 2nd licensed operator or other qualified individual is only allowed if ≥ 12 rods are withdrawn.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Tech Spec 3.3.2.1 required action C.2.2 allows the startup to commence. Plausible in that Tech Spec 3.3.2.1 action C.2.1.1 would allow an already in progress startup to continue if 12 rods were withdrawn.

B. Correct: OI 878.8, RWM Precaution and Limitation # 7 states that if, at any point, the RWM is found locked up, then reset the RWM by de-energizing and reenergizing its power per Section 6.2 of the procedure. Tech Spec 3.3.2.1 required actions C2.1.2 and C.2.2 allows rod withdraw if a startup has not been conducted within the last calendar year with the RWM inoperable and a 2nd licensed operator or other qualified individual is verifying control rod movement. Since the last plant startup was conducted on November 14, 2011 this requirement has been met.

C. Incorrect: OI 878.8, RWM Precaution and Limitation # 7 states that if, at any point, the RWM is found locked up, then reset the RWM by de-energizing and reenergizing its power per Section 6.2 of the procedure. Plausible if the candidate is unaware of this procedure requirement and the impact of placing the RWM Mode Switch in TEST.

D. Incorrect: OI 878.8, RWM Precaution and Limitation # 7 states that if, at any point, the RWM is found locked up, then reset the RWM by de-energizing and reenergizing its power per Section 6.2 of the procedure. Additionally, Tech Spec 3.3.2.1 required action C.2.2 allows the startup to commence.

Technical Reference(s):	OI 878.8, RWM Precaution and Limitation # 7	(Attach if not previously provided)
	Tech Spec 3.3.2.1	

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: N/A

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

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:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			1
	K/A #	G1	2.1.43	
	Importance Rating			4.3

Conduct of Operations: Ability to use procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, fuel depletion, etc.

Proposed Question: SRO Question # 99

Following a reduction in feedwater temperature, reactor power has been lowered using recirculation flow. The OATC has determined that load line will exceed 100.64%.

Which ONE of the following is required?

The CRS will direct…

A. recirculation flow be lowered as necessary to stay below the load line limit. A reactivity plan is required.

B. recirculation flow be lowered as necessary to stay below the load line limit. A reactivity plan is NOT required.

C. control rods be inserted as necessary to preclude an inadvertent violation of the load line limit. A reactivity plan is required.

D. control rods be inserted as necessary to preclude an inadvertent violation of the load line limit. A reactivity plan is NOT required.

Proposed Answer: D

Explanation (Optional):

A. Incorrect –Lowering recirculation flow will result in violating the load line limit. A reactivity plan is not required. Plausible because recirculation flow is used for fine adjustments of power and Reactivity plans are required for most power changes

B. Incorrect –Lowering recirculation flow will result in violating the load line limit. Plausible because recirculation flow is used for fine adjustments of power.

C. Incorrect –No reactivity plan is required. This action is necessary to preclude an inadvertent violation of the 100.4% admin limit due to limitations in monitoring load line during transients. Plausible because reactivity plans are required for most power changes.

D. Correct – If load line will exceed the load line limit (100.64%), then take immediate action to reduce power to the allowable region of the power to flow map with rods. No reactivity plan is required. This action is necessary to preclude an inadvertent violation of the 100.4% admin limit due to limitations in monitoring load line during transients. IAW IPOI 3, Reactivity plans are required for all planned core reactivity changes that will result in a change in power greater than or equal to 10 percent. Also During transient Xenon conditions action should be taken at an indicated load line of 100.0% to insert control rods. No reactivity plan is required. This action is necessary to preclude an inadvertent violation of the 100.4% admin limit due to limitations in monitoring load line during transients.

Technical Reference(s): AOP 646, pg 3
IPOI 3, P & L 10 & 35 (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: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.5
	Importance Rating		2.9

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

SRO Question # 100

An accident has occurred which has caused entry into the RPV Flood EOP. All control rods are fully inserted.

- Reactor water level is unknown
- RHR and Condensate are injecting into the reactor
- Reactor pressure is 400 psig and lowering quickly
- Torus water level is 10.4 feet and rising slowly
- Torus water temperature is 105°F and rising slowly
- All open SRV tailpipe temperatures are ~300°F and steady
- Drywell area Rad monitors are now reading 450 and 500 R/Hr and slowly rising
- Torus area Rad monitors are now reading 45 and 50 R/hr and slowly rising
- The TSC is NOT manned at this time

Using the Table below which ONE of the following actions is required at this time?

Table 5	**Core Damage Indications**
Parameter	Value
Primary containment hydrogen concentration	Drywell **OR** torus H_2 concentration above **0.4%** (minimum detectable)
Primary containment radiation	Drywell Area Hi Range Rad Monitor RIM-9184A/B above **7E+2 R/hr** OR Torus Area Hi Range Rad Monitor RIM-9185A/B above **3E+1 R/hr**
Reactor coolant activity	Chemistry samples above **300 µCi/gm** dose equivalent I-131
Fuel damage assessment (PASAP 7.2)	At or above **5%** fuel clad damage

A. Continue to perform RPV/F actions. SAG entry is not required until multiple Core damage indications are seen.

B. Continue to perform RPV/F actions, since core flooding indications exist, enter the SAGs and then exit the EOPs.

C. Continue to perform RPV/F actions until RPV reactor water level indications are observed. Once the TSC is operational, exit the EOPs and enter the SAGs.

D. Enter the SAGs and transition from RPV/F to SAG 1, Primary Containment Flooding. Continue to monitor the plant for degrading conditions and report them to the TSC once manned and operational.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – RPV/F actions must continue however there is a SAG entry condition on primary containment radiation. Plausible because RPV/F action must continue and the candidate may misinterpret Table 5.

B. Incorrect – EOPs should not be exited until there is evidence of core flooding and the TSC is operational. With reactor pressure still lowering and SRV tailpipe temperatures stable there is NO evidence of core flooding, with the core NOT flooded and the TSC NOT operational the EOPs should NOT be exited at this time. Plausible because there is an SAG entry condition on primary containment radiation.

C. Correct - Based upon the containment Rad levels during RPV flooding fuel damage is occurring. This requires entry to SAGs. The transition to SAGs is performed by continuing the EOP actions until there is evidence that the core is flooded or reactor water level indication is available and the TSC is operational. Once the TSC is operational, The SAGs are entered at the appropriate point and directed by the TSC and then the EOPs are exited and EOP actions terminated.

D. Incorrect – EOPs should NOT be exited until the TSC is operational. The transition to SAGs is performed by continuing the EOP actions until there is evidence that the core is flooded.

Technical Reference(s): RPV/F, Table 5
RPV/F Bases pg 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # DAEC Bank # QQ 20347

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content:	55.41	
	55.43	5

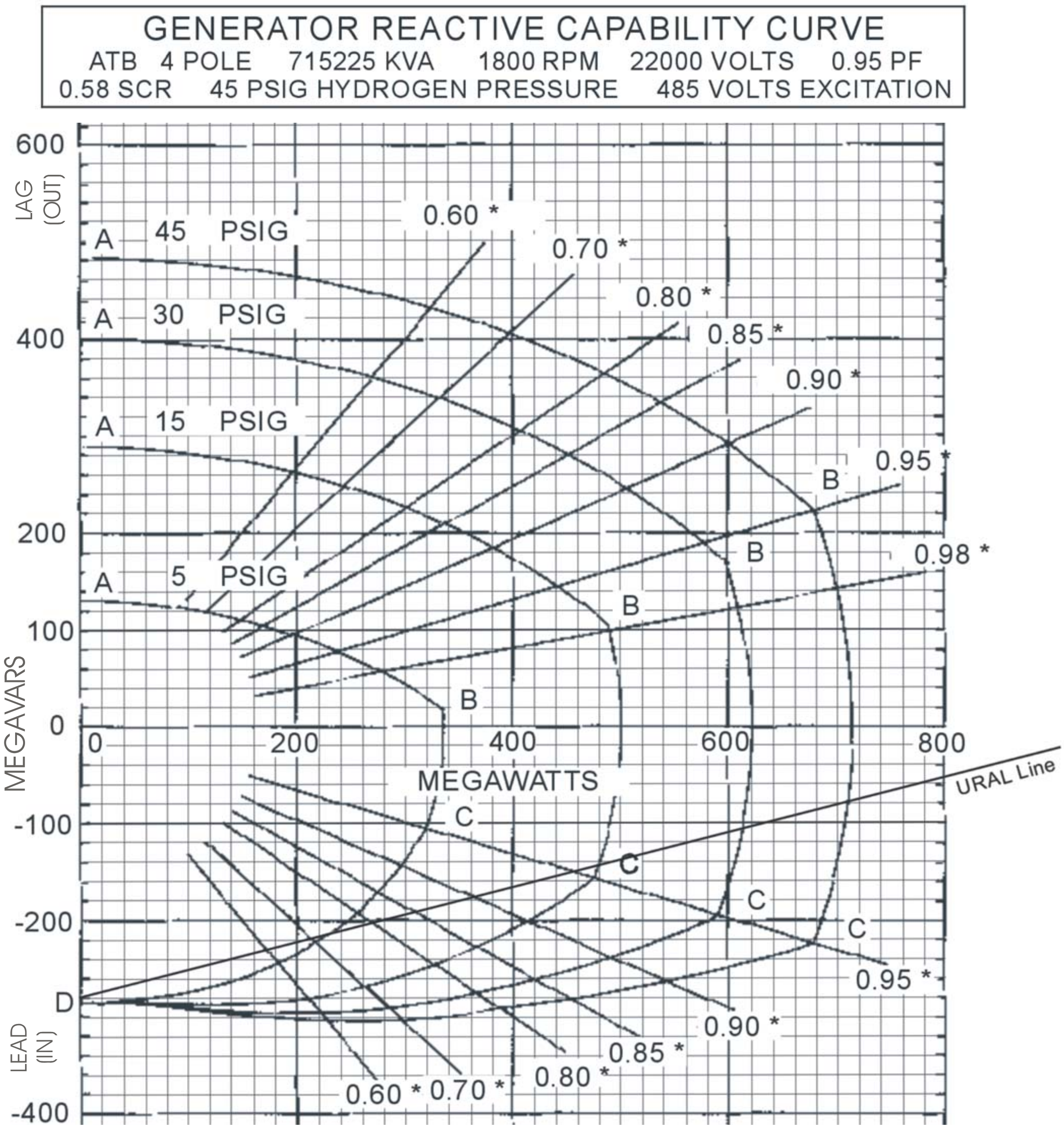
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

PDA 13-1 NRC Exam
Applicant References

1. OI 698 Attachment 1
2. AOP 149 Appendix 1&2
3. EOP Caution 1
4. EOP Graph 1
5. EOP Graph 5
6. EOP Graph 8
7. EOP Table 6
8. EAL-01 (Fission Product Barrier portion only)
9. L3.1.7
10. Figure 3.1.7-1
11. Figure 3.1.7-2
12. L 3.3.5.1
13. T3.3.5.1-1 (P1 thru 3, remove all allowable values)
14. L3.3.6.1
15. T3.3.6.1
16. L3.4.7
17. L3.4.8
18. L3.5.1
19. L3.6.4.1

APPENDIX 1
ESTIMATED CAPABILITY CURVES



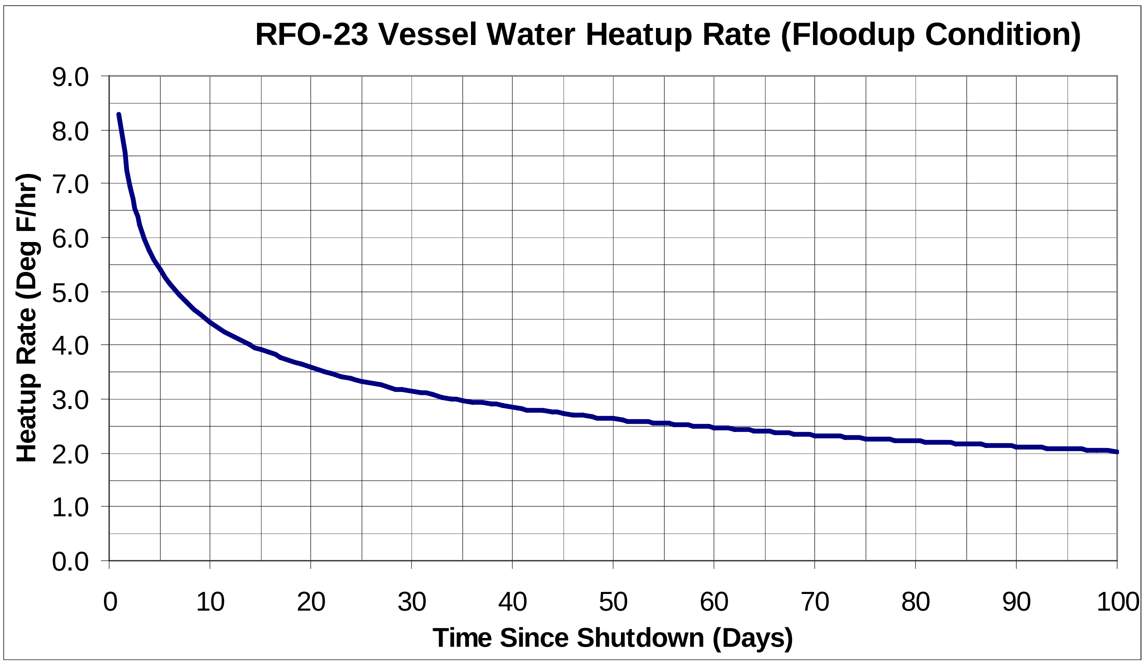
* PF (power factors)
CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

APPENDIX 1

HEATUP RATE CURVE - RPV FLOODED

NOTE

The RPV Flooded condition is defined as the RPV head removed, Spent Fuel Pool Gates removed, and the RPV and refuel cavity flooded up so that cavity level equals Spent Fuel Pool level. Spent Fuel Pool level is within the normal band.

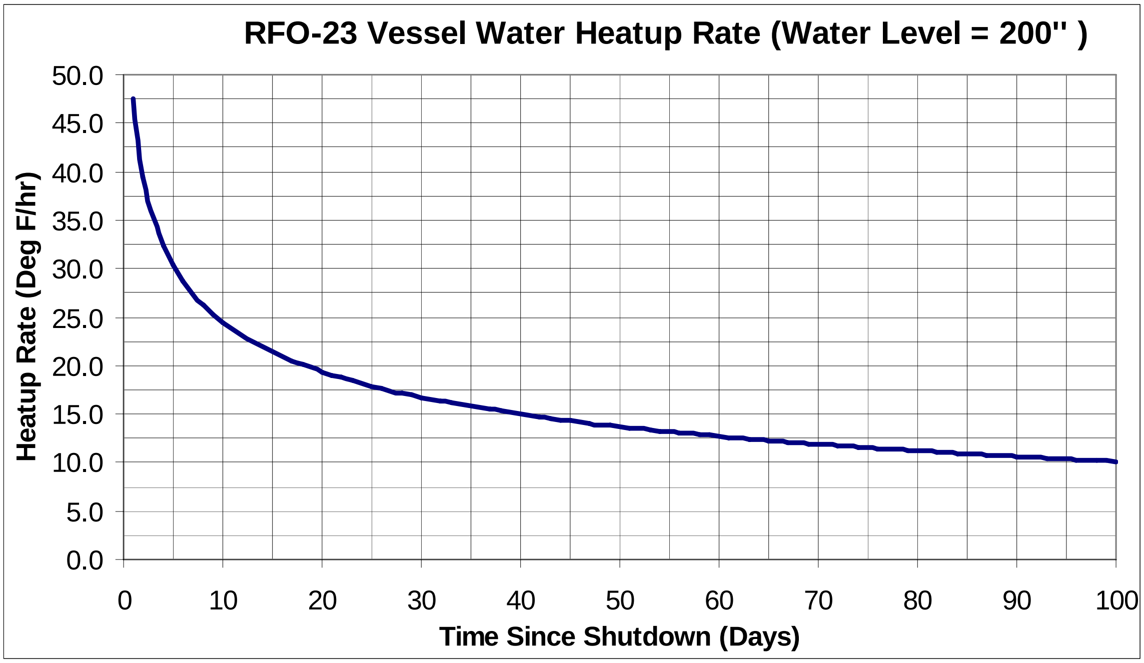


CAUTION

The initial heatup rate in the vessel may be higher than the calculated value when RHR or Fuel Pool Cooling is removed from service. The calculation used to generate the heatup rate curves assumes instantaneous mixing and heat transport from the fuel area to the remainder of the system volume. In addition, the calculated heatup rates reflect bulk temperatures not local temperatures. Under natural circulation conditions and the resulting time delay in heat transport, considerable differences in temperature may exist between the vessel and upper levels of the cavity or in the spent fuel pool. In some cases local boiling may occur but bulk boiling will not occur as long as cooling is restored within the calculated time-to-boil period.

APPENDIX 2

HEATUP RATE CURVE - RPV LEVEL AT 200"



CAUTION

The initial heatup rate in the vessel may be higher than the calculated value when cooling is removed from service. The calculation used to generate the heatup rate curves assumes instantaneous mixing and heat transport from the fuel area to the remainder of the system volume. In addition, the calculated heatup rates reflect bulk temperatures not local temperatures. Under natural circulation conditions and the resulting time delay in heat transport, considerable differences in temperature may exist between the fuel area and upper levels of vessel. In some cases local boiling may occur but bulk boiling will not occur as long as cooling is restored within the calculated time-to-boil period.

CAUTIONS

1

The following restrictions apply to RPV water level instruments:

1. If drywell air temperature is above the RPV Saturation Temperature (Graph 1), water in the instrument legs may boil. If boiling is suspected:
    a. Subtract 23 inches from Fuel Zone and Narrow Range GEMAC indications.
    b. Do not use Floodup and Wide Range Yarway instruments.
2. Floodup and Wide Range instruments may not be used below the Minimum Indicated Level for the indicated drywell temperature.

Level Instrument	Temperature Instrument	Drywell Temp (°F)	MIL (in.)
Wide Range Yarway LI-4539 (+8 to +218 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+8 +12 +16 +20 +25 +48
Wide Range Yarway LI-4540 (+8 to +218 in.)	TR-4383B Channel 2 (green)		
Floodup Range LI-4541 (+158 to +458 in.)	TR-4383A Channel 1 (red)	100-150 151-200 201-250 251-300 301-350 Upscale	+168 +174 +181 +189 +199 +257

2

Wide Range Yarways (LI-4539 and LI-4540) may be unreliable due to reference leg boiling during rapid RPV depressurization below 500 psig.

3

Operation of HPCI, RCIC, Core Spray, or RHR with suction from the torus and pump flow above the NPSH or vortex limit may damage equipment.

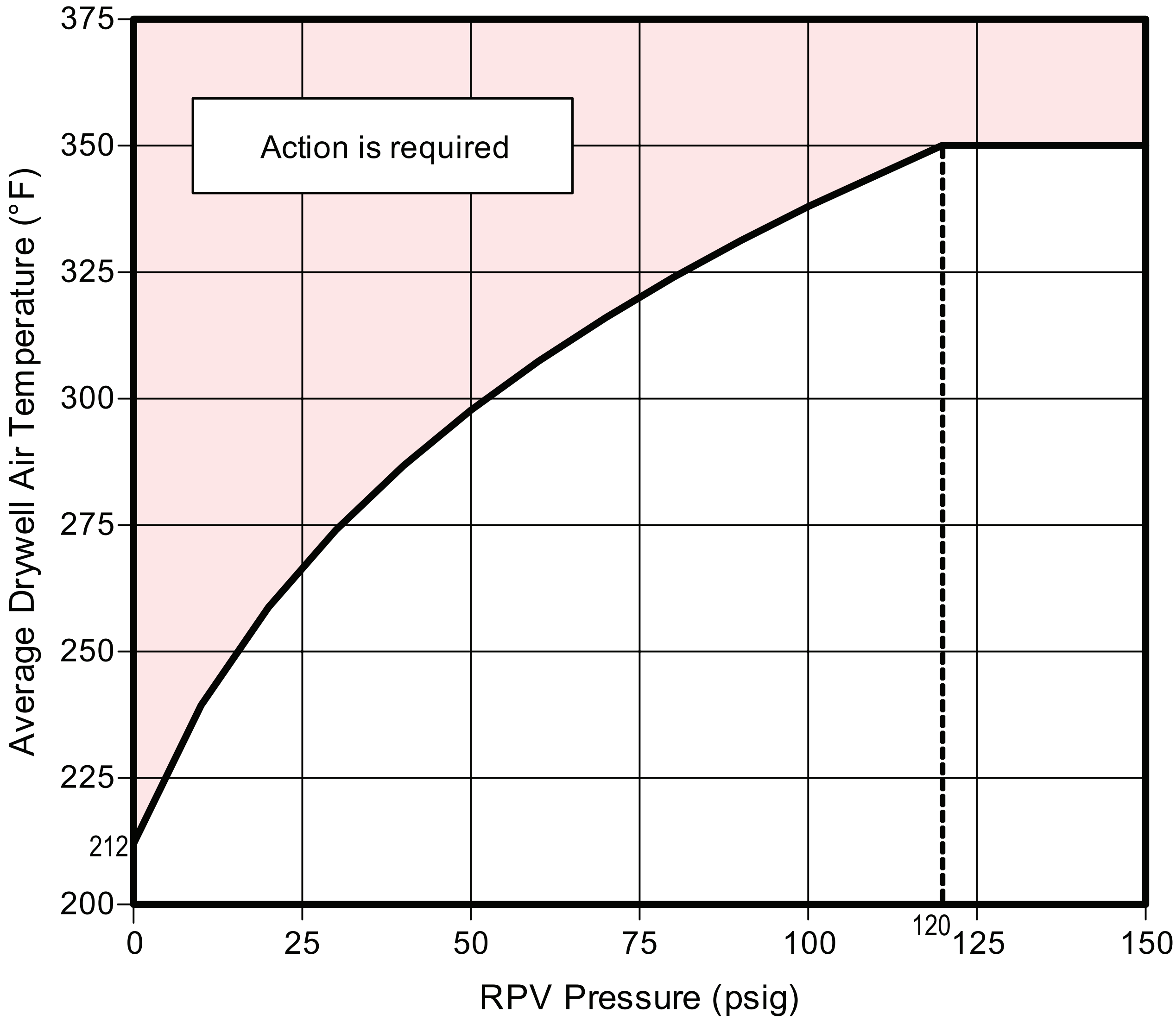
4

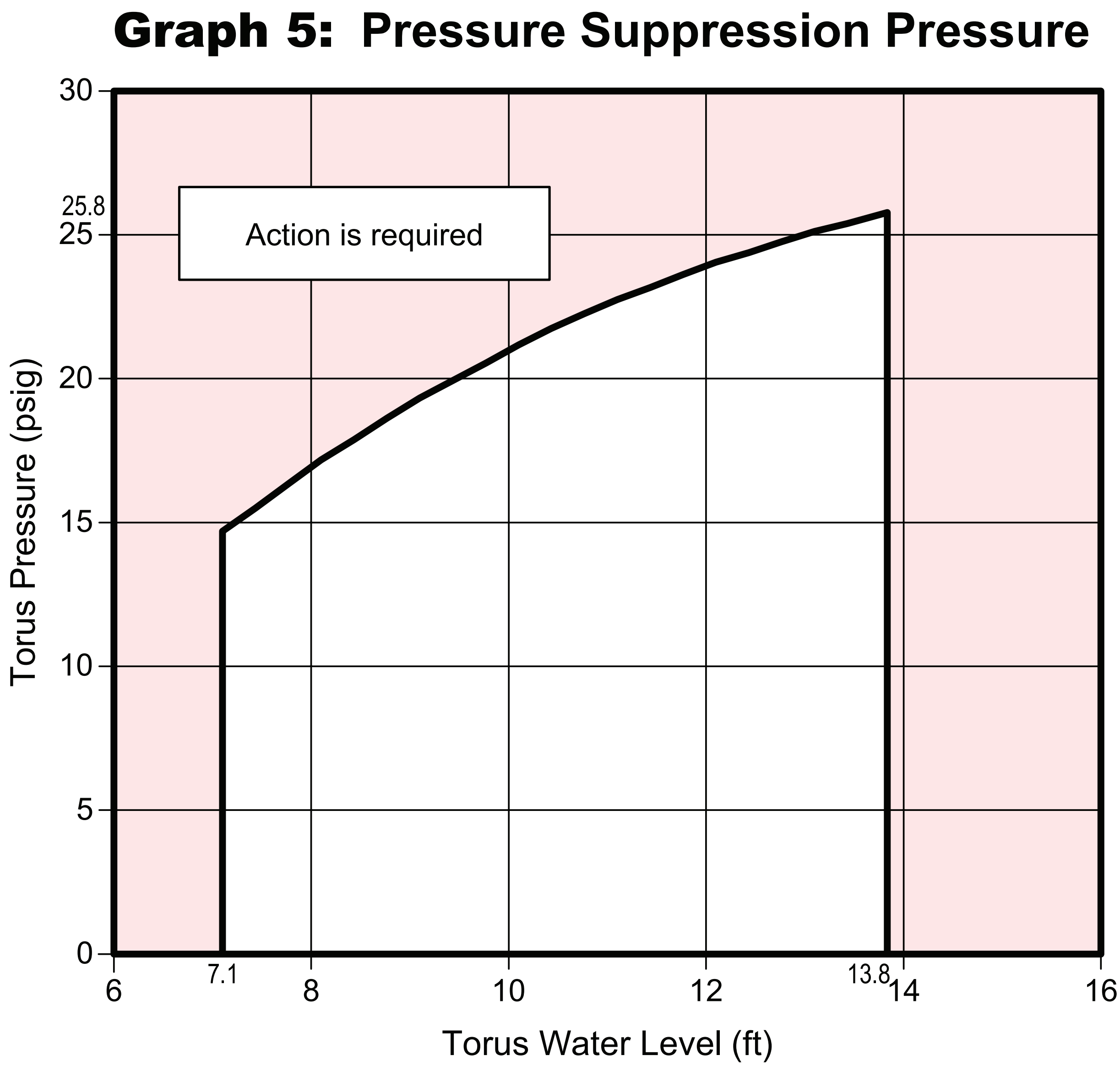
Elevated torus pressure may trip the RCIC turbine on high exhaust pressure.

8

Water may drain from isolated drywell cooling loops if well water is out of service when a PCIS Group 7 Isolation signal clears. This creates the potential for water hammer in drywell cooling piping when well water is restored.

GRAPH 1
RPV SATURATION TEMPERATURE




Graph 5: Pressure Suppression Pressure
Action is required
Torus Pressure (psig)
Torus Water Level (ft)
30
25.8
25
20
15
10
5
0
6
7.1
8
10
12
13.8
14
16

GRAPH 8
RHR NPSH

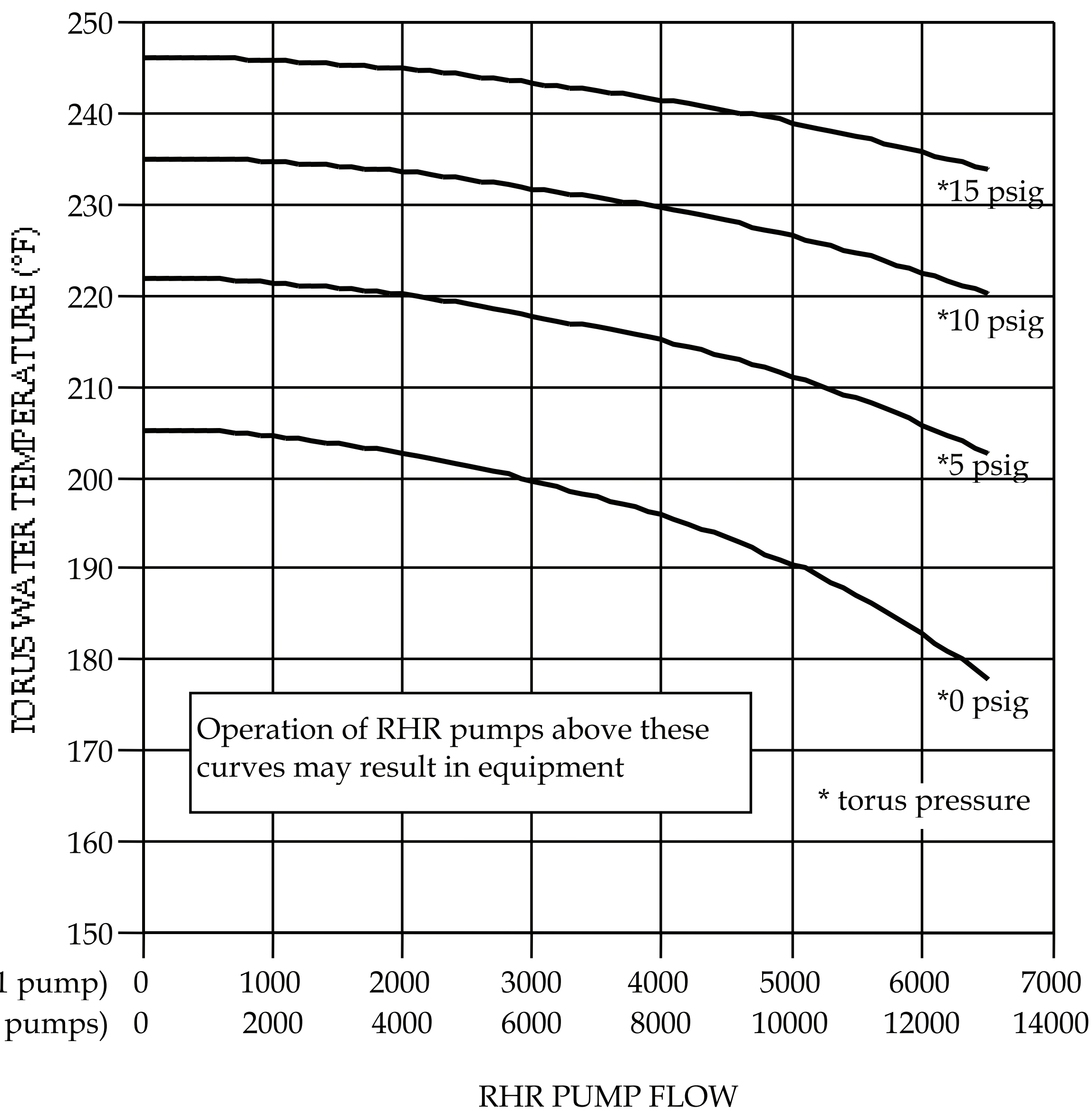


Table 6 Secondary Containment Limits

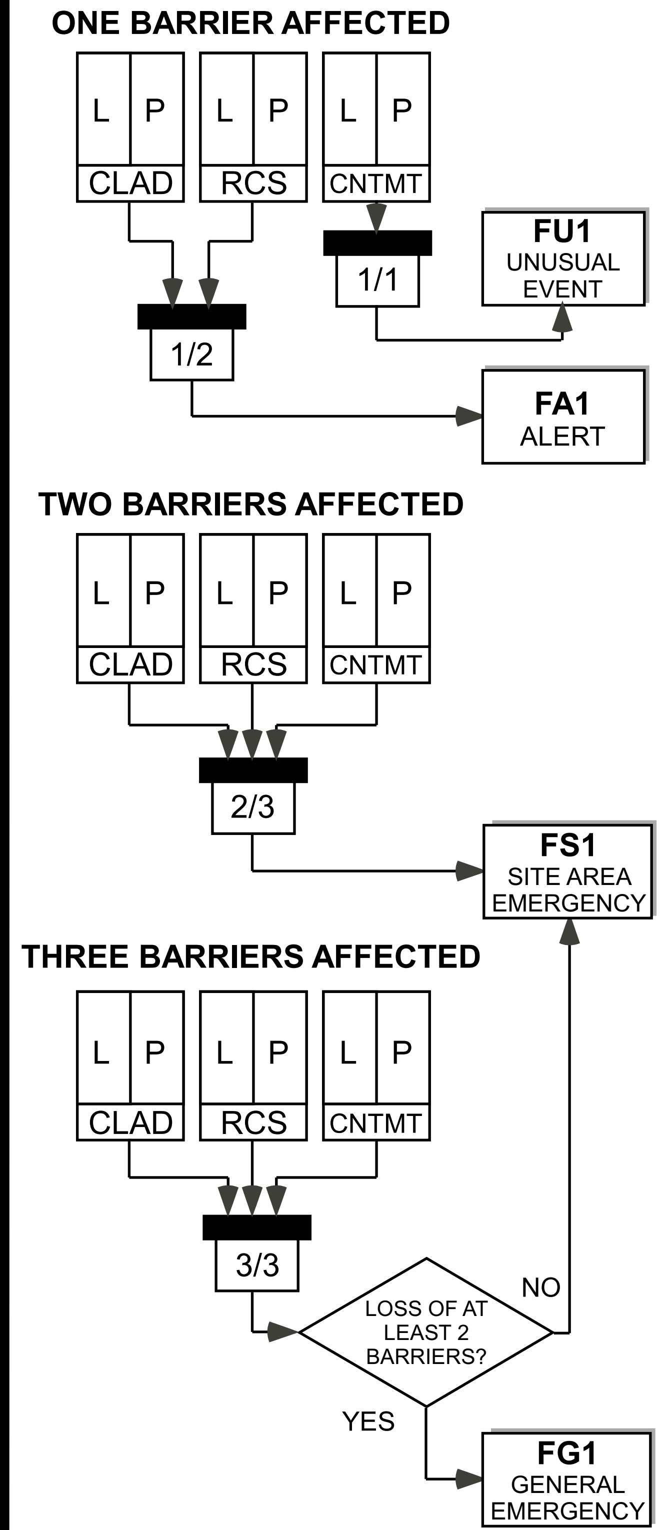
	Parameter and Areas		Max Normal Operating Limit	Max Safe Operating Limit	Value/Trend
Temperature	Area/Location	Indicator	°F	°F	
	A RHR–SE Corner Room Area				
	RHR SE CORNER ROOM AMBIENT	TR/TDR 2000A Ch 1	130	140	
	RHR SE CORNER ROOM DIFFERENTIAL	TR/TDR 2000A Ch 2	50	N/A	
	B RHR–NW Corner Room Area				
	RHR NW CORNER ROOM AMBIENT	TR/TDR 2000B Ch 1	130	140	
	RHR NW CORNER ROOM DIFFERENTIAL	TR/TDR 2000B Ch 2	50	N/A	
	HPCI Room Area				
	HPCI EMER COOLER AMBIENT	TR/TDR 2225A[B] Ch 1	175	310	
	HPCI ROOM AMBIENT	TR/TDR 2225A Ch 2	175	310	
	HPCI ROOM DIFFERENTIAL	TR/TDR 2225A[B] Ch 4[3]	50	N/A	
	RCIC Room Area				
	RCIC EMER COOLER AMBIENT	TR/TDR 2425A[B] Ch 1	175	300	
	RCIC ROOM AMBIENT	TR/TDR 2425A Ch 2	175	300	
	RCIC ROOM DIFFERENTIAL	TR/TDR 2425A[B] Ch 4	50	N/A	
	Torus Area				
	TORUS CATWALK NORTH AMBIENT	TR/TDR 2425A Ch 3	150	165	
	TORUS CATWALK WEST AMBIENT	TR/TDR 2425B Ch 2	150	165	
	TORUS CATWALK SOUTH AMBIENT	TR/TDR 2225A Ch 3	150	165	
	TORUS CATWALK EAST AMBIENT	TR/TDR 2225B Ch 2	150	165	
	TORUS CATWALK EAST DIFF	TR/TDR 2425A Ch 5	50	N/A	
	TORUS CATWALK WEST DIFF	TR/TDR 2425B Ch 5	50	N/A	
	TORUS CATWALK SOUTHWEST DIFF	TR/TDR 2225A Ch 5	50	N/A	
	TORUS CATWALK SOUTH DIFF	TR/TDR 2225B Ch 4	50	N/A	
	RB 786' South Area				
	RWCU PUMP ROOM AMBIENT	TR/TDR 2700A[B] Ch 1	130	212	
	RWCU HX ROOM AMBIENT	TR/TDR 2700A[B] Ch 2,3	130	212	
	RB 757' South Area				
	RWCU ABOVE TIP ROOM AMBIENT	TR/TDR 2700A[B] Ch 4,5	111.5	150	
	Steam Tunnel Area				
	STEAM TUNNEL AMBIENT	TR/TDR 2425B Ch 3	160	300	
	STEAM TUNNEL DIFFERENTIAL	TR/TDR 2225B Ch 5	70	N/A	
Radiation	Area/Location	Indicator	mR/hr	mR/hr	
	RB 757' South Area				
	RB RAILROAD ACCESS AREA	RI 9167	10	100	
	SOUTH CRD MODULE AREA	RI 9169	10	100	
	TIP ROOM	RI 9176	60	600	
	RB 757' North Area				
	NORTH CRD MODULE	RI 9168	10	100	
	CRD REPAIR ROOM	RI 9170	15	150	
	RB 786' North Area				
	RWCU SPENT RESIN ROOM	RI 9173	100	10^3	
	RWCU PHASE SEP TANK	RI 9177	20	200	
	RB 786' South Area				
	RWCU PUMP ROOM	RI 9156	10^3	10^4	
	RWCU HX ROOM	RI 9157	10^3	10^4	
	RB 812' North Area				
	MAIN PLANT EXHAUST FAN ROOM	RI 9171	60	600	
	JUNGLE ROOM	RI 9155	60	600	
	Refuel Floor Area				
	NEW FUEL VAULT AREA	RI 9153	10	100	
	NORTH REFUEL FLOOR	RI 9163	10	100	
	SOUTH REFUEL FLOOR	RI 9164	10	100	
	SPENT FUEL POOL AREA	RI 9178	100	10^3	
Water Level	Area/Location	Indicator	inches	inches	
	HPCI ROOM	LI 3768	2	6	
	RCIC ROOM	LI 3769	3	6	
	"A" RHR & CS SECR	LI 3770	2	10	
	"B" RHR & CS NWCR	LI 3771	2	10	
	TORUS AREA	LI 3772	2	12	

☐ FG1 1 2 3	☐ FS1 1 2 3	☐ FA1 1 2 3	☐ FU1 1 2 3
Loss of ANY Two Barriers AND Loss or Potential Loss of the Third Barrier (Table F-1)	Loss or Potential Loss of ANY Two Barriers (Table F-1)	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS Barrier (Table F-1)	ANY Loss or ANY Potential Loss of Primary Containment Barrier (Table F-1)

Fission Product Barriers

Table F-1 FISSION PRODUCT BARRIER MATRIX

Fuel Clad Barrier		RCS Barrier		Primary Containment Barrier	
☐ **Loss**	☐ **Potential Loss**	☐ **Loss**	☐ **Potential Loss**	☐ **Loss**	☐ **Potential Loss**
☐ RADIATION/CORE DAMAGE Fuel damage assessment (PASAP 7.2) indicates at least 5% fuel clad damage **OR** Drywell Area Hi Range Rad Monitor, RIM-9184A or B reading GREATER THAN 700 Rem/hr **OR** Torus Area Hi Range Rad Monitor, RIM-9185A or B reading GREATER THAN 30 Rem/hr **OR** Coolant activity GREATER THAN 300 µCi/gm DOSE EQUIVALENT I-131		☐ RADIATION/CORE DAMAGE Drywell Area Hi Range Rad Monitor RIM-9184A or B reading GREATER THAN 5 Rem/hr after reactor shutdown			☐ RADIATION/CORE DAMAGE Drywell Area Hi Range Rad Monitor, RIM-9184A or B reading GREATER THAN 3000 Rem/hr **OR** Torus Area Hi Range Rad Monitor, RIM-9185A or B reading GREATER THAN 100 Rem/hr **OR** Fuel damage assessment (PASAP 7.2) indicates at least 20% fuel clad damage
		☐ LEAKAGE UNISOLABLE Main Steamline, HPCI, Feedwater, RWCU or RCIC break as indicated by the failure of both isolation valves in any one line to close AND EITHER: • High MSL flow or high steam tunnel temperature annunciators • Direct report of steam release **OR** Emergency RPV Depressurization is required	☐ LEAKAGE RCS Leakage GREATER THAN 50 GPM inside the drywell **OR** UNISOLABLE primary system leakage outside the drywell as indicated by area temps or ARMs exceeding the Max Normal Limits per EOP 3, Table 6.	☐ LEAKAGE Failure of both isolation valves in any one line to close AND a direct downstream pathway to the environment exists after primary containment isolation signal. **OR** UNISOLABLE primary system leakage outside the drywell as indicated by area temps or ARMs exceeding the Max Safe Limits per EOP 3, Table 6, when Containment Isolation is required. **OR** Intentional Primary Containment venting per EOPs	
☐ RPV LEVEL RPV Level cannot be restored and maintained above -25 Inches	☐ RPV LEVEL RPV Level cannot be restored and maintained above +15 inches or cannot be determined	☐ RPV LEVEL RPV Level cannot be restored and maintained above +15 inches or cannot be determined			☐ RPV LEVEL Primary containment flooding required
		☐ PRIMARY CONTAINMENT ATMOSPHERE Drywell pressure GREATER THAN 2 psig due to RCS Leakage (not caused by a loss of DW Cooling)		☐ PRIMARY CONTAINMENT ATMOSPHERE Drywell pressure rise followed by a rapid unexplained drop in drywell pressure **OR** Drywell pressure response NOT consistent with LOCA conditions	☐ PRIMARY CONTAINMENT ATMOSPHERE Torus pressure reaches 53 psig and rising **OR** Drywell or Torus H_2 cannot be determined to be LESS THAN 6% and Drywell or Torus O_2 cannot be determined to be LESS THAN 5% **OR** RPV pressure and Torus water temperature cannot be maintained below the Heat Capacity Limit (EOP Graph 4)
☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the Fuel Clad Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the Fuel Clad Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the RCS Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the RCS Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the Containment Barrier	☐ EMERGENCY DIRECTOR JUDGMENT Any condition in the opinion of the Emergency Director (OSM, EC, or ER&RD) that indicates Loss or Potential Loss of the Containment Barrier



Modes: 1 Power Operation 2 Startup 3 Hot Shutdown 4 Cold Shutdown 5 Refueling DEF Defueled

Modes 1, 2, 3

Approved: Mike Davis, Manager Emergency Preparedness — 09/26/2012 Date

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.7.1	Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.3	Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.4	Verify continuity of explosive charge.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.5	Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 26.2 gpm at a discharge pressure ≥ 1150 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.8	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

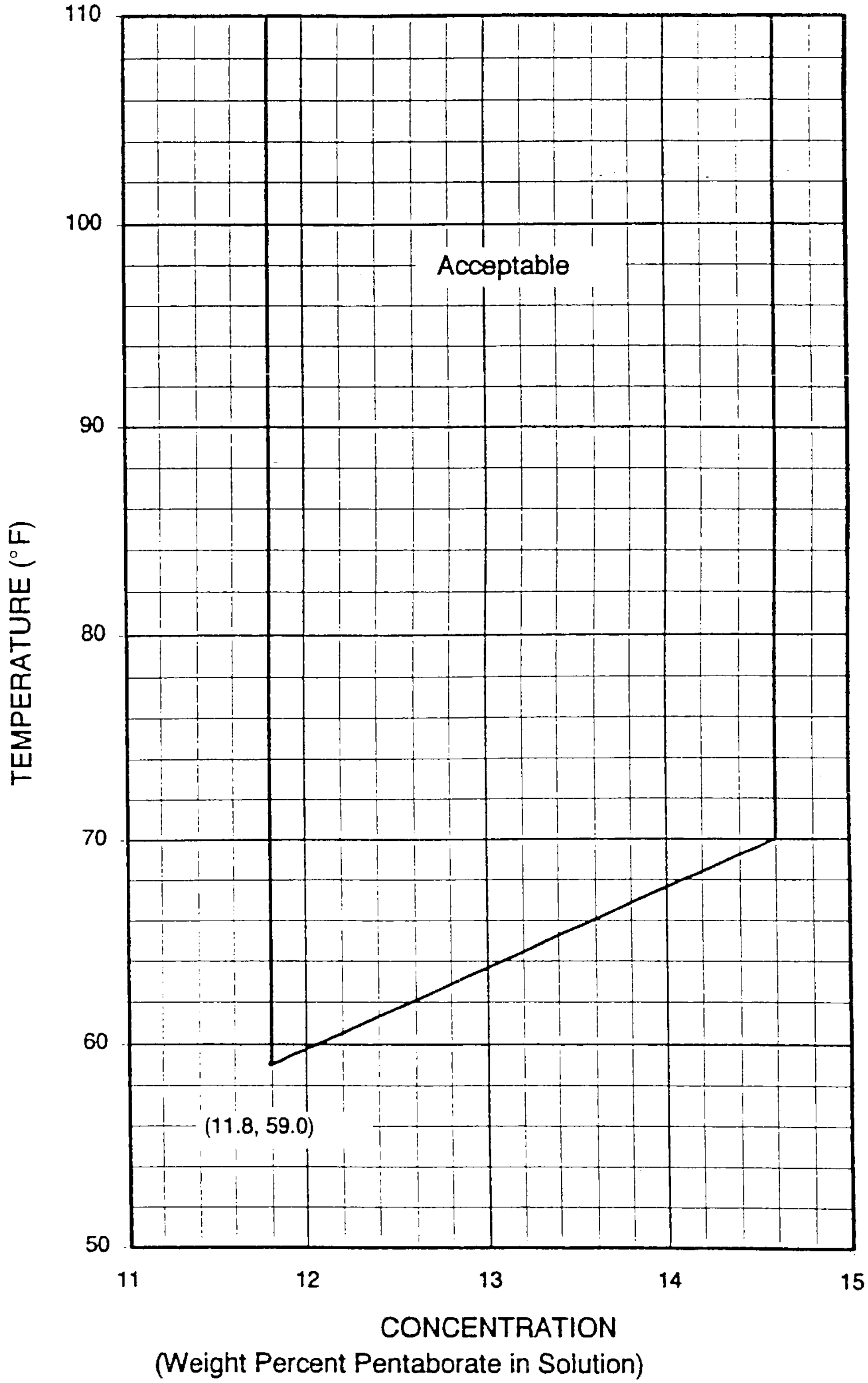


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

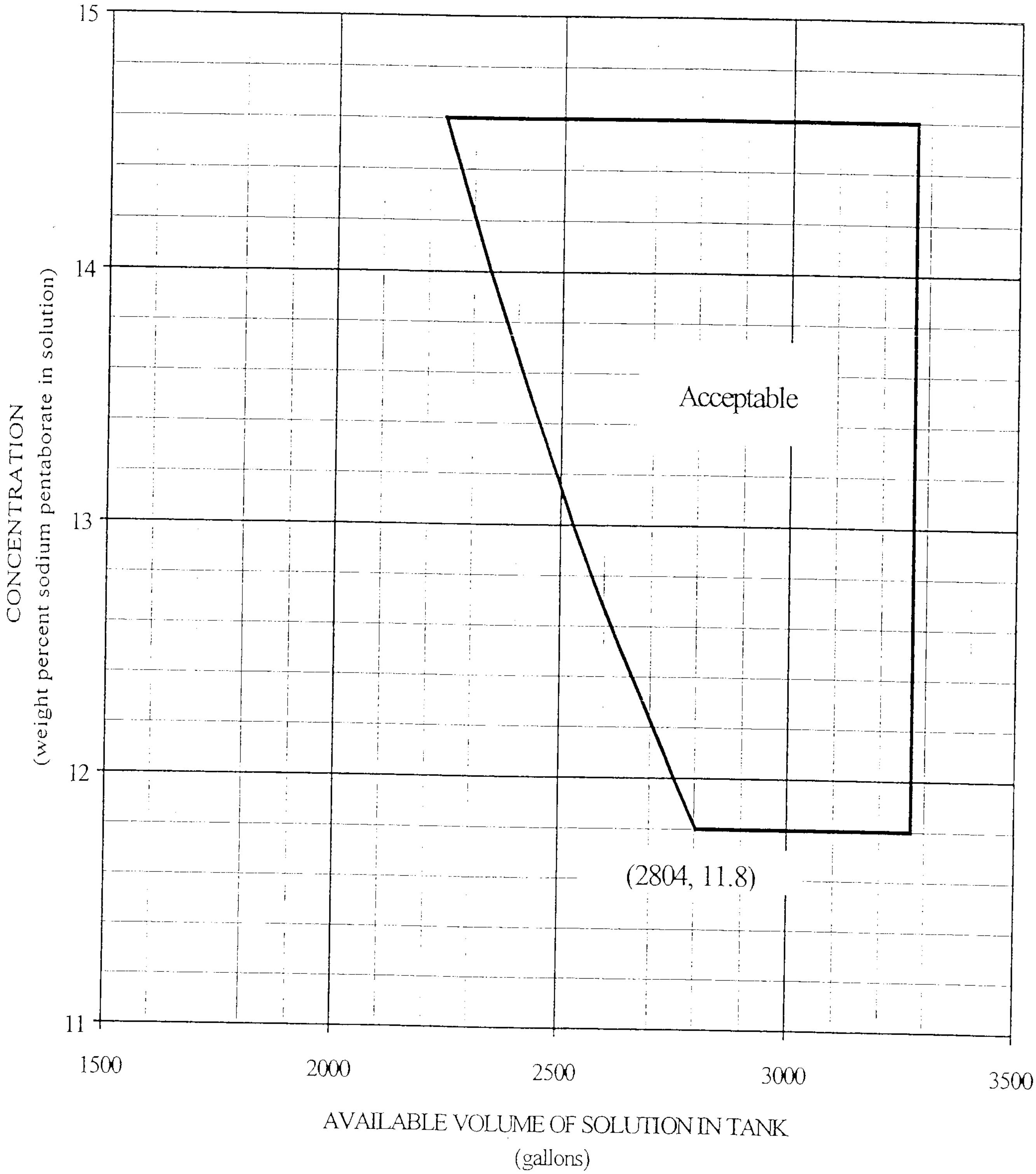


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

--NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1	B.1 --------------NOTES----------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. --------------------------------- Declare supported feature(s) inoperable when redundant feature(s) ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in two or more low pressure ECCS subsystems
	<u>AND</u>	
	B.2 --------------NOTE------------ Only applicable for Functions 3.a and 3.b. ---------------------------------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>	
	B.3 Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	------------NOTES---------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.e, 2.c and 2.e. -------------------------------- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for two or more low pressure ECCS subsystems
	<u>AND</u>		
	C.2	-------------NOTES-------- 1. Only applicable in Modes 1, 2, and 3. 2. Only applicable for Functions 2.g, 2.h, 2.i, and 2.j. ------------------------------- Declare Low Pressure Coolant Injection (LPCI) subsystem inoperable.	1 hour from discovery of loss of Loop Selection capability
	<u>AND</u>		
	C.3	Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1 ----------NOTE----------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ----------------------------- Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI suction transfer capability
	AND	
	D.2.1 Place channel in trip.	24 hours
	OR	
	D.2.2 Align the HPCI pump suction to the suppression pool.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1 ----------NOTES---------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.d and 2.f. ------------------------------- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for two or more minimum flow valves in the low pressure ECCS subsystems
	<u>AND</u>	
	E.2 Restore channel to OPERABLE status.	7 days
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1 Restore channel to OPERABLE status.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip logics
	AND	
	G.2 Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCI or Reactor Core Isolation Cooling (RCIC) inoperable AND 8 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	H.1 Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip logics
	AND	
	H.2 Restore channel to OEPRABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
I. Required Action and associated Completion Time of Condition B, C, D, E, F, G, or H not met.	I.1 Declare associated supported feature(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTES---

1. Refer to Table 3.3.5.1-1 to determine which SRs apply for each ECCS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Functions 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f; and (b) for up to 6 hours for Functions other than 1.d, 2.f, 3.c, 3.d, 3.e, and 3.f provided the associated Function (or the redundant Function for Functions 4 and 5) maintains ECCS initiation or loop selection capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.5.1.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.1.3	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.5.1.4	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.5.1.5	Perform CHANNEL FUNCTIONAL TEST.	12 months
SR 3.3.5.1.6	Perform CHANNEL CALIBRATION.	12 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.5.1.7	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.5.1.8	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.5.1.9	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low	1,2,3, 4[(a)], 5[(a)]	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
b. Drywell Pressure - High	1,2,3	4[(b)]	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
c. Reactor Steam Dome Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
	4[(a)], 5[(a)]	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
d. Core Spray Pump Discharge Flow – Low (Bypass)	1,2,3, 4[(a)], 5[(a)]	1 per pump	E	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
e. Core Spray Pump Start Time Delay Relay	1,2,3, 4[(a)], 5[(a)]	1 per pump	C	SR 3.3.5.1.8 SR 3.3.5.1.9	
f. 4.16 kV Emergency Bus Sequential Loading Relay	1,2,3, 4[(a)], 5[(a)]	1 per pump	F	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.9	
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level- Low Low Low	1,2,3, 4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
b. Drywell Pressure - High	1,2,3	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS-Shutdown.

(b) Also required to initiate the associated Diesel Generator (DG).

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
c. Reactor Steam Dome Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
	$4^{(a)}, 5^{(a)}$	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
d. Reactor Vessel Shroud Level - Low	1,2,3	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.9	
e. Low Pressure Coolant Injection Pump Start - Time Delay Relay	1,2,3, $4^{(a)}, 5^{(a)}$	1 per pump	C	SR 3.3.5.1.8 SR 3.3.5.1.9	
Pumps A & B					
Pumps C & D					
f. Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1,2,3, $4^{(a)}, 5^{(a)}$	1 per loop	E	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
g. LPCI Loop Select- Reactor Vessel Water Level - Low-Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.6 SR 3.3.5.1.9	
h. LPCI Loop Select – Reactor Steam Dome Pressure - Low	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.9	

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS – Shutdown.

Table 3.3.5.1-1 (page 3 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
i. LPCI Loop Select – Recirculation Pump Differential Pressure	1,2,3	4 per pump	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.8 SR 3.3.5.1.9	
j. LPCI Loop Select – Recirculation Riser Differential Pressure	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.9	
k. 4.16 kV Emergency Bus Sequential Loading Relay	1,2,3	2	F	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.9	
	$4^{(a)}$, $5^{(a)}$	1	F	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.9	
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low	1, $2^{(c)}$, $3^{(c)}$	4	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.6 SR 3.3.5.1.9	
b. Drywell Pressure - High	1, $2^{(c)}$, $3^{(c)}$	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	
c. Reactor Vessel Water Level - High	1, $2^{(c)}$, $3^{(c)}$	2	C	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.6 SR 3.3.5.1.9	
d. Condensate Storage Tank Level - Low	1, $2^{(c)}$, $3^{(c)}$	2	D	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	

(a) When the associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS – Shutdown.

(c) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.b, 6.b, and 6.c <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.b, and 6.b, and 6.c
	<u>AND</u>	
	A.2 ----------NOTE--------- Only applicable for Function 7.a. --------------------------- Inhibit containment spray system.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more automatic Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. [Deleted]		
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	12 hours 36 hours
I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	I.1 Declare Standby Liquid Control (SLC) System inoperable. OR I.2 Isolate the Reactor Water Cleanup System.	1 hour 1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	J.1 Initiate action to restore channel to OPERABLE status.	Immediately
	OR	
	J.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System.	Immediately
K. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	K.1 -----------NOTE------------ Only applicable if inoperable channel is not in trip. ------------------------------	
	Declare associated Suppression Pool Cooling/Spray subsystem(s) inoperable.	Immediately
	OR	
	K.2 ----------NOTE------------ Only applicable if inoperable channel is in trip. ------------------------------	
	Declare Primary Containment inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
L. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	L.1 Isolate the primary containment vent and purge penetration flow paths.	1 hour
	<u>OR</u>	
	L.2 Establish administrative control of the primary containment vent and purge valves using continuous monitoring of alternate instrumentation.	1 hour

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Primary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed as follows: (a) for up to 6 hours for Function 5.a; and (b) for up to 6 hours for Functions other than 5.a provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.2	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.3	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.4	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.6	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.7	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.8	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.6.1.9	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\geq$ 38.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	$\geq$ 821 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	$\leq$ 138% rated steam flow
d. Condenser Backpressure - High	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\geq$ 7.2 inches Hg vacuum
e. Main Steam Line Tunnel Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	$\leq$ 205.1°F
f.. Turbine Building Temperature - High	1,2,3	4	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	$\leq$ 205.1°F

(continued)

(a) When any turbine stop valve is greater than 90% open or when the key-locked bypass switch is in the NORM Position.

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\geq$ 165.6 inches
b. Drywell Pressure - High	1,2,3	2	H	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 2.2 psig
c. Offgas Vent Stack - High Radiation	1(c), 2(c), 3(c)	1	L	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	(b)
d. Reactor Building Exhaust Shaft – High Radiation	1,2,3	1	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 12.8 mR/hr
e. Refueling Floor Exhaust Duct – High Radiation	1,2,3	1	H	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 10.6 mR/hr
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	$\leq$ 409 inches (inboard) $\leq$ 110 inches (outboard)

(continued)

(b) Allowable value is determined in accordance with the ODAM.

(c) During venting or purging of primary containment.

Table 3.3.6.1-1 (page 3 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
b. HPCI Steam Supply Line Pressure – Low	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 50 psig and ≤ 147.1 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 2.5 psig
d. Drywell Pressure - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
e. Suppression Pool Area Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 153.3°F
f. HPCI Leak Detection Time Delay	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	N/A
g. Suppression Pool Area Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F
h. HPCI Equipment Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 178.3°F
i. HPCI Room Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F

(continued)

Table 3.3.6.1-1 (page 4 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 164 inches (inboard) ≤ 159 inches (outboard)
b. RCIC Steam Supply Line Pressure - Low	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 50.3 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.9	≥ 3.3 psig
d. Drywell Pressure – High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
e. RCIC Suppression Pool Area Ambient Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 153.3°F
f. RCIC Leak Detection Time Delay	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	N/A
g. RCIC Suppression Pool Area Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F
h. RCIC Equipment Room Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 178.3°F
i. RCIC Room Ventilation Differential Temperature - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 51.5°F

(continued)

Table 3.3.6.1-1 (page 5 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 59 gpm
b. Area Temperature - High	1,2,3	1(d)	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 133.3°F
c. Area Ventilation Differential Temperature – High	1,2,3	1(d)	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	
RWCU Pump Room RWCU Pump A Room RWCU Pump B Room RWCU Heat Exch. Room					≤ 22.5°F ≤ 23.5°F ≤ 34.5°F ≤ 51.5°F
d. SLC System Initiation	1,2	1(e)	I	SR 3.3.6.1.9	NA
e. Reactor Vessel Water Level – Low Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≥ 112.65 inches
f. Area Near TIP Room Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 115.7°F
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.9	≤ 152.7 psig
b. Reactor Vessel Water Level – Low	3,4,5	2(f)	J	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 165.6 inches
c. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.4 SR 3.3.6.1.8 SR 3.3.6.1.9	≤ 2.2 psig
7. Containment Cooling System Isolation					
a. Containment Pressure – High	1,2,3	4	K	SR 3.3.6.1.3 SR 3.3.6.1.8 SR 3.3.6.1.9	≥ 1.25 psig

(d) Each Trip System must have either an OPERABLE Function 5.b or an OPERABLE Function 5.c channel in both the RWCU pump area and in the RWCU heat exchanger area.

(e) SLC System Initiation only inputs into one of the two trip systems.

(f) Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown

LCO 3.4.7 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----------------------------------NOTES--

1. Both required RHR shutdown cooling subsystems and recirculation pumps may not be in operation for up to 2 hours per 8 hour period.

2. One required RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 3, with reactor steam dome pressure < the RCIC Steam Supply Line Pressure – Low isolation pressure.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore required RHR shutdown cooling subsystem(s) to OPERABLE status. <u>AND</u>	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify by administrative means an alternate method of decay heat removal is available for each required inoperable RHR shutdown cooling subsystem. AND A.3 Be in MODE 4.	1 hour 24 hours
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation. AND B.2 Verify reactor coolant circulation by an alternate method. AND B.3 Monitor reactor coolant temperature and pressure.	Immediately 1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	---------------------------NOTE--------------------------- Not required to be met until 2 hours after reactor steam dome pressure is < the RCIC Steam Supply Line Pressure – Low isolation pressure. -- Verify one required RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

--------------------------------------NOTES-------------------------------------

1. Both required RHR shutdown cooling subsystems and recirculation pumps may not be in operation for up to 2 hours per 8 hour period.

2. One required RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify by administrative means an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND	
	B.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one required RHR shutdown cooling subsystem or one recirculation pump is operating.	12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS — Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of four safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except High Pressure Coolant Injection (HPCI) is not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 100 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Residual Heat Removal (RHR) pump inoperable.	A.1 Restore RHR pump to OPERABLE status.	30 Days
B. One low pressure ECCS subsystem inoperable for reasons other than Condition A.	B.1 Restore low pressure ECCS subsystem to OPERABLE status.	7 days
C. One Core Spray subsystem inoperable. AND One or two RHR pump(s) inoperable.	C.1 Restore Core Spray subsystem to OPERABLE status. OR C.2 Restore RHR pump(s) to OPERABLE status.	72 hours 72 hours
D. Both Core Spray subsystems inoperable.	D.1 Restore one Core Spray subsystem to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
F. HPCI System inoperable.	F.1 Verify by administrative means RCIC System is OPERABLE. AND F.2 Restore HPCI System to OPERABLE status.	Immediately 14 days
G. HPCI System inoperable. AND One RHR pump inoperable.	G.1 Restore HPCI System to OPERABLE status. OR G.2 Restore RHR pump to OPERABLE status.	7 days 7 days
H. HPCI System inoperable. AND One low pressure ECCS subsystem is inoperable for reasons other than Condition A.	H.1 Restore HPCI System to OPERABLE status. OR H.2 Restore low pressure ECCS subsystem to OPERABLE status.	72 hours 72 hours
I. HPCI System inoperable. AND One ADS valve inoperable.	I.1 Restore HPCI System to OPERABLE status. OR I.2 Restore ADS valve to OPERABLE status.	72 hours 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. Required Action and associated Completion Time of Condition F, G, H, or I not met.	J.1 Be in MODE 3. <u>AND</u> J.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
K. One ADS valve inoperable.	K.1 Restore ADS valve to OPERABLE status.	30 days
L. One ADS valve inoperable. <u>AND</u> One low pressure ECCS subsystem inoperable for reasons other than Condition A.	L.1 Restore ADS valve to OPERABLE status. <u>OR</u> L.2 Restore low pressure ECCS subsystem to OPERABLE status.	72 hours 72 hours
M. Two or more ADS valves inoperable. <u>OR</u> Required Action and associated Completion Time of Condition K or L not met.	M.1 Be in MODE 3. <u>AND</u> M.2 Reduce reactor steam dome pressure to ≤ 100 psig.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. Two or more low pressure ECCS subsystems inoperable for reasons other than Condition C or D. <u>OR</u> HPCI System and two or more ADS valves inoperable. <u>OR</u> HPCI System and two or more low pressure ECCS subsystems inoperable. <u>OR</u> One ADS valve and two or more low pressure ECCS subsystems inoperable. <u>OR</u> One ADS valve and HPCI System and one low pressure ECCS subsystem inoperable.	N.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.2	-------------------------------NOTE------------------------------------ The low pressure coolant injection (LPCI) system may be considered OPERABLE during alignment and operation for decay heat removal in MODE 3, if capable of being manually realigned and not otherwise inoperable. --- Verify each ECCS injection/spray subsystem power operated and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify a 100 day supply of nitrogen exists for each ADS accumulator.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.4	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF Core Spray / ≥ 2718 gpm / 1 / ≥ 113 psig LPCI / ≥ 4320 gpm / 1 / ≥ 20 psig	In accordance with the Inservice Testing Program

SYSTEM	FLOW RATE	NO. OF PUMPS	SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF
Core Spray	≥ 2718 gpm	1	≥ 113 psig
LPCI	≥ 4320 gpm	1	≥ 20 psig

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1025 and ≥ 940 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.	In accordance with the Inservice Testing Program
SR 3.5.1.6	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 160 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.7	-----------------------------NOTES----------------------------- 1. Vessel injection /spray may be excluded. 2. For the LPCI System, the Surveillance may be met by any series of sequential and/or overlapping steps, such that the LPCI Loop Select function is tested. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	-------------------------NOTE------------------------------- Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.9	Verify each ADS valve actuator strokes when manually actuated.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During Operations with a Potential for Draining the Reactor Vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Secondary containment inoperable during OPDRVs.	C.1 -----------NOTE---------- LCO 3.0.3 is not applicable. ----------------------------- Initiate action to suspend OPDRVs.	 Immediately

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify all secondary containment equipment hatches are closed.	31 days
SR 3.6.4.1.2	-----------------------NOTE--------------------------- Doors in high radiation areas may be verified by administrative means. --- Verify that either the outer door(s) or the inner door(s) in each secondary containment access opening are closed.	 31 days
SR 3.6.4.1.3	Verify each SBGT subsystem can maintain $\geq$ 0.25 inch of vacuum water gauge in the secondary containment at a flow rate $\leq$ 4000 cfm.	24 months on a STAGGERED TEST BASIS