Susquehanna - December 2007
Senior Reactor Operator
NRC Written Exam Answer Key

1.	B	26.	C	51.	B	76.	D
2.	C	27.	B	52.	B	77.	A
3.	D	28.	C	53.	A	78.	C
4.	A	29.	B	54.	B	79.	D
5.	A	30.	C	55.	B	80.	B
6.	D	31.	B	56.	B	81.	A
7.	A	32.	B	57.	D	82.	A
8.	C	33.	B	58.	B	83.	C
9.	C	34.	B	59.	C	84.	C
10.	D	35.	B	60.	D	85.	C
11.	C	36.	A	61.	D	86.	B
12.	D	37.	C	62.	D	87.	B
13.	B	38.	B	63.	A	88.	B
14.	D	39.	D	64.	C	89.	A
15.	C	40.	A	65.	D	90.	D
16.	B	41.	D	66.	C	91.	D
17.	D	42.	C	67.	C	92.	C
18.	A	43.	B	68.	C	93.	B
19.	D	44.	A	69.	C	94.	D
20.	A	45.	A	70.	C	95.	B
21.	A	46.	C	71.	C	96.	B
22.	A	47.	D	72.	D	97.	D
23.	D	48.	D	73.	B	98.	D
24.	B	49.	C	74.	B	99.	A
25.	A	50.	B	75.	B	100.	B

Susquehanna - December 2007
Reactor Operator
NRC Written Exam Answer Key

1.	B	26.	C	51.	B
2.	C	27.	B	52.	B
3.	D	28.	C	53.	A
4.	A	29.	B	54.	B
5.	A	30.	C	55.	B
6.	D	31.	B	56.	B
7.	A	32.	B	57.	D
8.	C	33.	B	58.	B
9.	C	34.	B	59.	C
10.	D	35.	B	60.	D
11.	C	36.	A	61.	D
12.	D	37.	C	62.	D
13.	B	38.	B	63.	A
14.	D	39.	D	64.	C
15.	C	40.	A	65.	D
16.	B	41.	D	66.	C
17.	D	42.	C	67.	C
18.	A	43.	B	68.	C
19.	D	44.	A	69.	C
20.	A	45.	A	70.	C
21.	A	46.	C	71.	C
22.	A	47.	D	72.	D
23.	D	48.	D	73.	B
24.	B	49.	C	74.	B
25.	A	50.	B	75.	B

FINAL Submittal

SSES
Reactor Operator
NRC
Written Exam

Questions
1 - 75

Exam Date: 12/7/2007

FINAL Submittal

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 K5.01	
	Importance Rating	2.7	

K5.01 - Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) : Testable check valve operation

Proposed Question: Common 1

During RHR Cold Shutdown Valve Exercising surveillance testing, the RHR Testable Check valve HV-151-F050A failed to actuate. Maintenance reports that SV-15150A did not energize when the pushbutton was depressed. The SRO declared the valve inoperable.

Which ONE of the following describes the effect this would have on the testable check valve AND the RHR system?

A. Instrument Air (IA) would not be admitted to the check valve actuating cylinder. The operation of the testable check valve during a LPCI initiation would not be assured.

B. Instrument Gas (CIG) would not be admitted to the check valve actuating cylinder. The operation of the testable check valve during a LPCI initiation would not be assured.

C. Instrument Air (IA) would not be admitted to the check valve actuating cylinder. The ability to prevent flow from the reactor recirculation loop to the RHR system would not be assured.

D. Instrument Gas (CIG) would not be admitted to the check valve actuating cylinder. The ability to prevent flow from the reactor recirculation loop to the RHR system would not be assured.

Proposed Answer: B

Explanation (Optional):

A. Incorrect- Instrument Gas is used to actuate the cylinder.
B. Correct
C. Incorrect – Instrument Gas is used to actuate the cylinder. The testable checks will continue to prevent back flow from the reactor recirculation loop into the RHR System.
D. Incorrect - The testable check will continue to prevent back flow from the reactor recirculation loop into the RHR System

Per TM-OP-049-ST, Rev.3

Check Valves (HV-151-F050A and B) located in the Primary Containment, act to prevent back flow from the reactor recirculation loop into the RHR System preventing loss of inventory from the RPV and preventing overpressurization of the RHR piping. The valves are located in each loop between the F015 and F060 Valves. **To ensure the check valves will operate when required for LPCI,** the valves are equipped with an actuating cylinder that operates the disc for testing. The actuating cylinder is attached to a lever arm that moves the check valve disc into the open position. The lever arm also engages limit switches for red-open/amber- closed indication on 1C601 above the control switch.

Spring Return Pushbutton (RHR LOOP A/B TESTABLE CKV HV-151-FO5OA/B) on 1C601 energizes two solenoid valves when depressed (releasing the pushbutton de-energizes the solenoids). **The first Solenoid, SV-15150A/B admits Instrument Gas to the check valve-actuating cylinder.** The second Solenoid, SV-15122A/B admits Instrument Gas to the pneumatic operator of Valve HV-151-F122A/B (which is the bypass around the FO5OA/B Valve). Bypassing around the disc of F050A/B permits pressure to equalize across the valve, thereby minimizing the force required to open the check valve disc. Amber-closed/red-open position indication is provided on 1C601.

Both sets of A and B solenoids receive power from 1Y216 via Panel 1C613.

Technical Reference(s): TM-OP-049-ST, Rev.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K2.02	
	Importance Rating	2.5	

K2.02 - Knowledge of electrical power supplies to the following: Motor operated valves (shutdown cooling)

Proposed Question: Common 2

Unit 1 is operating in Shutdown Cooling with the "A" RHR pump in service when a loss of power occurs on MCC 1D274 "Reactor Building 250 VDC HPCI Control Center".

Which ONE of the following describes which valves are affected by this power loss?

A. HV-151-F008, RHR Shutdown Cooling Header Outboard Isolation.
HV-151-F009, RHR Shutdown Cooling Header Inboard Isolation.

B. HV-151-F009, RHR Shutdown Cooling Header Inboard Isolation.
HV-151-F006A, RHR Pump "A" Shutdown Cooling Suction Valve.

C. HV-151-F008, RHR Shutdown Cooling Header Outboard Isolation.
HV-151-F023, Reactor Vessel Head Spray Outboard Isolation.

D. HV-151-F009, RHR Shutdown Cooling Header Inboard Isolation.
HV-151-F022, Head Spray Shutoff Inboard Isolation Valve.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – F009 is AC powered
B. Incorrect – F006A and F009 are AC powered
C. Correct
D. Incorrect – F009 is AC powered and F022 is AC powered from 1B237.

Per TM-OP-049-ST, Rev.3, Page 55

The 250 VDC Power System provides power to the motors and control circuits of the following valves in the RHR System:

- HV-151-F008, Shutdown Cooling Header Isolation
- HV-151-F023, Reactor Vessel Head Spray Outboard Isolation
- HV-151-F049, RHR System Drain to LRW Outboard Isolation

Each of these valves has a corresponding inboard isolation valve that is AC powered. The valves close automatically upon receipt of a Primary Containment Isolation signal.

Technical Reference(s): TM-OP-049-ST, Rev.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K2.03	
	Importance Rating	2.8	

Knowledge of electrical power supplies to the following: Initiation Logic: BWR-2,3,4

Proposed Question: Common 3

Unit 1 is operating at 100% power with the following conditions:

- 125V DC PANEL 1L620 SYSTEM TROUBLE ALARM (AR-106 B12)
- The field operator reports that "125V DC DIST PNL 1D624 is de-energized."
- A valid HPCI initiation signal is then received.

Which ONE of the following describes the impact on the HPCI system?

A. HPCI will not automatically initiate because the HPCI Aux Oil pump has lost power.

B. HPCI must be manually initiated by arming and depressing the HPCI MAN INIT HS-E41-1S33 because both HPCI instrument logic channels have lost power.

C. HPCI will automatically initiate and inject because one logic channel for HPCI is still powered.

D. HPCI must be manually initiated component by component because only the initiation logic channel has lost power.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Aux oil pump is powered from 250VDC.
B. Incorrect - HPCI initiation logic has only one power supply.
C. Incorrect – HPCI will not auto initiate without 1D624 logic power.
D. Correct

Per TM-OP-052, Rev 1, page 55

HPCI INITIATION LOGIC

The HPCI Initiation Logic consists of a single channel, powered from 125 VDC (1D624), and can be actuated either automatically or manually.

Technical Reference(s): TM-OP-052, Rev 1, page 55 (Attach if not previously provided)
ON-102-620, Loss of 125V DC BUS 1D620

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K4.02	
	Importance Rating	3.0	

Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Prevents water hammer

Proposed Question: Common 4

A loss of Condensate Transfer has occurred.

Which ONE of the following describes how water hammer would be prevented in the Core Spray system?

The Passive Keepfill Head Tank is normally aligned to supply_______.

A. the pump discharge piping just upstream of the outboard injection valves full of water for at least 8 hours.

B. the pump discharge piping between the outboard and inboard injection valves full of water for at least 8 hours.

C. the pump discharge piping just upstream of the outboard injection valves full of water for a maximum of 4 hours.

D. the pump discharge piping between the outboard and inboard injection valves full of water for a maximum of 4 hours.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – the piping is aligned upstream of the outboard injection valve. This would be correct for RHR
C. Incorrect – the piping is aligned upstream of the outboard injection valve for at least 8 hours.
D. Incorrect - the piping is aligned upstream of the outboard injection valve (this would be correct for RHR) for at least 8 hours.

Per TM-OP-051-ST, Rev.2, Pages 45 & 46.

The Discharge Line Fill System minimizes the time lag between a pump start signal and the initiation of flow into the RPV, because it keeps the pump discharge piping of the respective system full of water. **Simultaneously, it also prevents water hammer during the rapid start transient of the ECCS pumps.** Should the discharge piping not be filled and pressurized, water hammer effects could result in unnecessary shock to system piping and components, which could result in component failure.

The Discharge Line Fill System headers to the CS System provide a continuous supply of water from the Condensate Storage and Transfer System to the CS loop injection headers just upstream of the outboard injection valves, and each header contains the following components (Figure 1):

- A pressure reducing station consisting of a one-inch, manually operated, Globe Valve (152-F025A(B)), a Flow Indicator (FI-15226A(B)), a Pressure Control Valve (PCV-152-F026A(B)), and a one-inch, manually operated, Globe Valve (152-F027A(B))
- A two-inch, manually operated, normally locked open, pressure reducing station Bypass Valve (152-F028A(B))
- One-inch Pressure Relief Valve (PSV-152-F024A(B))
- Two two-inch In-Series Check Valves (152-F029A(B), 152-F030A(B))

The Discharge Line Fill System supply header to both CS loop injection headers is provided with a Relief Valve (PSV-152-F024A(B)) located downstream of the pressure reducing station bypass valve in the Reactor Building on Elevation 683', Area 25. These valves function to prevent overpressurization of the Discharge Line Fill System piping in the event leakage occurs across the in-series check valves and CS loop inboard injection valves. Each valve is a one-inch spring-loaded relief valve with a capacity of 10 gpm at a lift pressure of 180 psig. The relief valves discharge to the floor on Elevation 683'.

The backup, passive, ECCS Keepfill Tank will keep HPCI, RCIC, Core Spray, and RHR filled for greater than eight hours upon loss of the pressurized source from a Condensate Transfer Pump.

ON-037-001, Rev.7, discussion

Technical Reference(s): TM-OP-051-ST, Rev.2 (Attach if not previously provided)

ON-037-001, Rev.7, discussion

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K1.06	
	Importance Rating	3.7	

Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Reactor vessel

Proposed Question: Common 5

Given the following conditions on Unit 1:

- Control rods failed to insert on a scram.
- Initial ATWS power is 100%.
- Standby Liquid Control (SLC) was initiated however one pump tripped.
- The running 1A SLC pump is operating properly within design limits.
- MSIVs are open.

Which ONE of the following is assured with one SLC pump injecting?

A. The injection rate will be quick enough to compensate for Xenon decay and a cooldown rate of 100°/hr.

B. The injection rate ensures that the Heat Capacity Temperature Limit of the Suppression Pool during a 100 percent ATWS will not be violated.

C. flow through the system will have sufficient velocity, so no plate out of SLC mixture in the injection line and proper mixing of the boron mixture in the core occurs.

D. sufficient time will remain for refill of the SLC tank after the initial hot shutdown boron volume was injected.

Proposed Answer: A

Explanation (Optional):

A. Correct To compensate for xenon decay and 100 °F/hr cooldown rate.

B. Incorrect - HCTL will be violated regardless of born injection rate with a 100% ATWS

C. Incorrect - 41.2 gpm is not a basis for boron velocity and mixing in regard to plate out per the TS bases.

D. Incorrect - Refill time for SLC tank is not an ATWS concern per TS bases.

Per TS Unit 1 bases 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate > 41.2 gpm at a discharge pressure > 1395 psig without actuating the pump's relief valve ensures that pump performance has not degraded during the fuel cycle. Testing at 1395 psig assures that the functional capability of the SLC system meets the ATWS Rule (10 CFR 50.62) (Ref. 1) requirements. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay.

Technical Reference(s): TS bases SR 3.1.7.7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP053/10096

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.01	
	Importance Rating	2.7	

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Fuel thermal time constant

Proposed Question: Common 6

Which ONE of the following describes how the Neutron Monitoring System inputs to the Reactor Protection System function to protect the reactor?

A. For rapid neutron flux increase events, the neutron flux lags the thermal power and the APRM Fixed Neutron Flux—High Function will provide a scram signal after the APRM Flow Biased Simulated Thermal Power—High Function setpoint is exceeded.

B. For loss of feedwater heating events, the thermal power significantly lags the neutron flux and the APRM Fixed Neutron Flux—High Function will provide a scram signal before the APRM Flow Biased Simulated Thermal Power—High Function setpoint is exceeded.

C. The APRM Flow Biased Simulated Thermal Power—High Function monitors neutron flux and is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the thermal power in the reactor. The reactor trip is clamped at an upper limit that is always HIGHER than the APRM Fixed Neutron Flux—High Function Allowable Value.

D. The APRM Flow biased Simulated Thermal Power—High Function monitors neutron flux and is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the thermal power in the reactor. The reactor trip is clamped at an upper limit that is always LOWER than the APRM Fixed Neutron Flux—High Function Allowable Value.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - For rapid neutron flux increase events, the thermal power lags the neutron flux
B. Incorrect – For loss of FW heating events thermal power does not significantly lag neutron flux
C. Incorrect - The reactor trip is clamped at an upper limit that is always LOWER than the Average Power Range Monitor Neutron Flux—High Function Allowable Value.
D. Correct

Per Unit 1 TS Bases 3.3.1.1.2.b

The Average Power Range Monitor Simulated Thermal Power—High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux—High Function Allowable Value. The Average Power Range Monitor Simulated Thermal Power—High Function is not credited in any plant Safety Analyses. The Average Power Range Monitor Simulated Thermal Power – High Function is set above the APRM Rod Block to provide defense in depth to the APRM Neutron Flux – High for transients where THERMAL POWER increases slowly (such as loss of feedwater heating event). During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Neutron Flux—High Function will provide a scram signal before the Average

Technical Reference(s): Unit 1 TS Bases 3.3.1.1.2.b (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K6.05	
	Importance Rating	3.1	

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

Proposed Question: Common 7

Unit 1 reactor mode switch is in Startup when an IRM instrument spiked upscale then returned to its previous value.

Which ONE of the following describes how the control room determines which IRM spiked?

A. The IRM Trip Unit seals in on the affected IRM drawer in the relay room.

B. The IRM CHAN UPSCALE TRIP OR INOP alarm on 1C651 remains in alarm.

C. 1C651 IRM benchboard UPSC TR OR INOP red lamp light seals in.

D. Powerplex core monitoring computer printout lists the most recent alarms.

Proposed Answer: A

Explanation (Optional):

A	Correct.
B	Incorrect. Once a condition clears reset will clear the alarm
C	Incorrect. Benchboard lamp lit only when condition exists. Will not seal in.
D	Incorrect. Core monitoring computer printout does not list the most recent alarms.

Technical Reference(s): TM-OP-078B-ST, Rev.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #	TMOP078/17/17	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

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

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

Proposed Question: Common 8

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

- IRM A: 25 on Range 2.
- Reactor mode switch: STARTUP

While selecting SRM detectors for withdrawal to maintain SRM Countrate between 1 X $10E^3$ to 1 X $10E^5$ CPS, IRM A is also selected. The drive out pushbutton is then depressed and held depressed.

Which statement describes the expected plant response to this event?

A. The IRM "A" detector will not withdraw because the Reactor Mode Switch is in Startup.

B. When the IRM "A" full-in limit switch opens, the IRM inop circuit will generate a 1/2 scram.

C. A rod block occurs when the IRM "A" full-in limit switch opens but no 1/2 scram occurs.

D. Both a rod block and a 1/2 scram occurs when the IRM "A" full-in limit switch opens.

Proposed Answer: C

Explanation (Optional):

A	Incorrect; a rod block occurs
B	Incorrect; this does not cause an inop condition
C	Correct; the retract permit
D	Incorrect; only a rod block occurs due to this condition

AR-104-H03, Rod Out Block, Probable Cause 1.3.8 IRM Detector Position Wrong and Page 14 of TM-OP-078B-ST, IRM Rev 3.

Technical Reference(s): AR-104-H03 (Attach if not previously provided)
TM-OP-078B-ST

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP078B/C
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A1.05	
	Importance Rating	3.6	

Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: SCRAM, rod block, and period alarm trip setpoints

Proposed Question: Common 9

During a refueling outage on Unit 2, the RPS Shorting Links have been removed.

Which ONE of the following will result if the "A" Source Range Monitor (SRM) drawer mode switch is taken out of the OPERATE position?

A. A SRM downscale alarm.
No Rod Block or Reactor Scram will occur.

B. A SRM downscale alarm
A Rod Block will occur.

C. A SRM Upscale/Inop alarm.
A Rod Block will occur.

D. A SRM Upscale/Inop alarm.
A Rod Block and a Half Scram will occur.

Proposed Answer: C

Explanation (Optional):

A Incorrect; placing the mode switch out of operate generates an INOP rod block but not a downscale alarm.
B Incorrect; A downscale alarm will not occur.
C Correct;
D Incorrect; only an upscale condition inputs to the reactor scram circuitry. No half scram will occur. Out of operate will cause an Upscale/Inop alarm and a Rod Block

Per TM-OP-078A-ST, Rev.2

Fact Sheet:

SRM Trips and associated Setpoints:
• INOP condition from any of the following:
High voltage low
Mode switch not in OPERATE
Module unplugged
Trip is bypassed when all IRM channels for that division are on RANGE 8 or above, or the Reactor Mode Switch is in RUN, or the Joystick is in Bypass.
Protection Trip Signals
- Reactor Scram signal generated from Upscale Trip

Control Rod Withdrawal Block generated from:
- Upscale Alarm

INOP
- Downscale
- Retract Permissive is not satisfied

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A reactor scram can be generated by an SRM Upscale Trip signal if the RPS Shorting Links are removed.

The SRM inop contacts are in series with the SRM upscale contacts in the rod block circuit, but only the upscale contacts are present in the scram circuit. Placing the SRM mode switch out of operate will cause the SRM inop trip and resultant rod block but there will be no input to the scram circuit. The inop trip (due to mode switch position) will only cause a rod block and upscale/inop alarm.

Technical Reference(s): TM-OP-078A-ST, Rev.2 (Attach if not previously provided)
SI-178-215A, Rev.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP078A/1345
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A1.04	
	Importance Rating	4.1	

Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: SCRAM and rod block trip setpoints

Proposed Question: Common 10

The plant is operating at 65 percent power with the OPRMs enabled. The following alarms are received:

"OPRM TRIP" received from APRM 1 (TRIP active on ODA)
"APRM UPSCALE OR INOP TRIP" received from APRM 4 (TRIP active on ODA)

Which ONE of the following is the automatic plant response?

A. A HALF RPS Scram. A FULL RPS Scram does NOT occur.

B. A FULL RPS Scram

C. A Rod Block and FULL RPS Scram

D. Rod Block and NO RPS Scram

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Two Votes are present
B. Incorrect. One OPRM Vote and One APRM Vote is NO Scram
C. Incorrect. Two Votes are present
D. Correct - If one APRM Channel trips and one OPRM Channel trips, a reactor scram will not result because the APRM 2/4 logic and the OPRM 2/4 logic are independent of each other. The rod block logic only requires one trip input for the rod block.

Technical Reference(s): TM-OP-78D-ST, Rev.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # TMOP078D/15710

Modified Bank # ____________ (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	217000 A2.12	
Importance Rating	3.0	

Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings

Proposed Question: Common 11

The RCIC system is in operation with an initiation signal present. A loss of the Topaz inverter occurs.

Which ONE of the following describes the RCIC response and what actions are required to control RCIC flow?

A. The Governor valve fails as-is.
Use the HV-150-F045 "Turbine Steam Admission Valve" to control turbine speed.

B. The Governor valve fails as-is.
Use the turbine trip and throttle valve HV-15012 to control turbine speed.

C. The Governor valve fails open.
Use the turbine trip and throttle valve HV-15012 to control turbine speed.

D. The Governor valve fails open.
Use the HV-150-F045 "Turbine Steam Admission Valve" to control turbine speed.

Proposed Answer: C

Explanation:

A. Incorrect – The governor valve would fail open. HV-150-F045 "Turbine Steam Admission Valve" cannot be throttled since it has an auto open signal due to the initiation signal.
B. Incorrect – The governor valve would fail open.
C. Correct – Procedure directs using the turbine trip and throttle valve HV-15012 to control turbine speed.
D. Incorrect. - HV-150-F045 "Turbine Steam Admission Valve" cannot be throttled since it has an auto open signal due to the initiation signal.

TM-OP-050-ST, Rev 04
<u>Page 45</u>
A failure of the inverter will result in the loss of the Turbine Control System and the Turbine Governor (Control) Valve (FV-15013) failing to the "FULL OPEN" position. System operation is still possible under these conditions by manually controlling the position of the Turbine Trip and Throttle (Stop) Valve (HV-15012). Operation under these conditions should be limited until full automatic control can be restored.

Technical Reference(s): TM-OP-050-ST, Rev.04 (Attach if not previously provided)
OP-150-001, Rev.27

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K1.04	
	Importance Rating	2.6	

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

Proposed Question: Common 12

Which ONE of the following RCIC failures would result in Main Condenser air inleakage?

A. With RCIC running, an air leak on the RCIC barometric condenser.

B. With RCIC running, an air leak on the RCIC steam line drain header.

C. With RCIC in standby, an air leak on the RCIC barometric condenser.

D. With RCIC in standby, an air leak on the RCIC steam line drain header.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – The drains are routed via the main steam line header to the main condenser while in standby.

B. Incorrect – The drains are routed via the main steam line header to the main condenser while in standby.

C. Incorrect - The drains are routed via the main steam line header to the main condenser while in standby.

D. Correct

Per TM-OP-050-ST, Rev.4

The main condenser receives steam drains from the Turbine Steam Supply Drain Header Level Control System (via the main steam line drain headers) when the RCIC System is in the "Standby Mode." The use of the main condenser as a collection point permits returning the condensate to the RPV without affecting storage inventory levels. During RCIC System operation, the drain header is isolated, and any condensation formed will be carried with the steam flow and exhausted to the Suppression Pool.

Technical Reference(s): TM-OP-050-ST, Rev.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 2.2.25	
	Importance Rating	2.5	

Equipment Control Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (Automatic Depressurization System)

Proposed Question: Common 13

Which ONE of the following describes the bases of the Technical Specification surveillance requirement (TS 3.5.1.3) that every 31 days the ADS gas supply header pressure must be verified to be ≥ 135 psig?

Following a failure of the pneumatic supply (CIG) to ADS SRVs, accumulators are sized such that ____________.

A. at least TWO valve actuations can occur with the drywell at 70% of design pressure.

B. at least ONE valve actuation can occur with the drywell at 70% of design pressure.

C. at least TWO valve actuations can occur with the drywell at 85% of design pressure.

D. at least ONE valve actuation can occur with the drywell at 85% of design pressure.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – only ONE actuation is required.
B. Correct.
C. Incorrect – only ONE actuation is required at 70% design drywell pressure.
D. Incorrect – the actuation is based on 70% design drywell pressure.

Per bases 3.5.1.3

Verification every 31 days that ADS gas supply header pressure is ≥ 135 psig ensures adequate gas pressure for reliable ADS operation. 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, at least one valve actuations can occur with the drywell at 70% of design pressure.

Technical Reference(s): TS Bases 3.5.1.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: Per TM-OP-083E-OB, LO Systems, Automatic Depressurization System

12702 - Demonstrate knowledge of the Technical Specifications and Technical Requirements Bases associated with the Automatic Depressurization System for Background, Applicable Safety Analyses, LCO, Applicability, and Surveillance Requirements.

Question Source:

Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Level:

Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content:

55.41 X

55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A2.10	
	Importance Rating	3.9	

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 abn cond or ops. Loss of coolant accidents

Proposed Question: Common 14

A Unit 1 reactor scram has occurred. The following conditions exist:

- RPV Level lowered to +1 inch, and is currently +18 inches and steady.
- Drywell pressure reached 2.1 psig, and is currently 0.75 psig and lowering.
- In accordance with ON-159-002, "Containment Isolation", the MN STM LINE DIV 1 and DIV 2 ISO RESET Pushbuttons (HS-B21-1S32 and HS-B21-1S33) have been depressed.

Based on your actions above to this point, which of the following describes components that will be completely reset, i.e., ready to align and startup the system?

A. RBCW to DW coolers
CRM Sample Valves

B. CAC H_2/O_2 Analyzers
Drywell Equipment Drain Valves

C. RBCW Supply to RRP Motor Cooler Valves
Drywell Floor Drain Valves

D. CAC H_2/O_2 Analyzers
CRM Sample Valves

Proposed Answer: D

Explanation (Optional):

A. Incorrect. RBCW also requires reset pushbuttons depressed on Panel 1C601, DW Cooling Logic.
B. Incorrect. Equipment drains also have additional reset pushbuttons on Panel 1C601.
C. Incorrect. RBCW also requires additional reset pushbuttons depressed on back panel, 1C681, DW Cooling Logic. DW Floor Drain Valves have separate reset pushbuttons.
D. Correct.

Per ON-159-02, Rev.26

Steps 5.0 and 6.0 for RBCW
Step 7.0 for CAC System (no reset required, just open valves)
Step 8.0 for CRM system (no reset required, just open valves)
Step 9.0 for Equipment drains

Technical Reference(s): ON-159-002, Rev.26 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP059B/10481
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K6.03	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF : Process radiation monitoring system

Proposed Question: Common 15

During 100 percent power operation on Unit 1, the "B" RPS MG Set fails, resulting in a loss of the "B" RPS Bus.

What is the response of the Containment Radiation Monitoring System (CRMs) Containment Isolation Valves to this failure?

A. The Inboard Isolation Valves close, and the Outboard Isolation Valves remain as-is in both "A" and "B" CRMs.

B. The Outboard Isolation Valves close, and the Inboard Isolation Valves remain as-is in both "A" and "B" CRMs.

C. The Outboard and Inboard Isolation Valves close in both "A" and "B" CRMs.

D. The Outboard and Inboard Isolation Valves close in the "B" CRMs; the "A" CRM Isolation Valves remain as-is.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - A loss of RPS B initiates inboard and outboard containment isolation valves.
B. Incorrect - A loss of RPS B initiates inboard and outboard containment isolation valves.
C. Correct
D. Incorrect - Each DIV isolation affects both CRM loops.

ON-158-001, Loss of RPS, Rev.11

Technical Reference(s): ON-158-001, Loss of RPS, Rev.11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # TMOP079X/1903

Modified Bank # ______________ (Note changes or attach parent)

New ______________

Question History: Last NRC Exam ______________

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K3.03	
	Importance Rating	4.3	

Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following: Ability to rapidly depressurize the reactor.

Proposed Question: Common 16

On Unit 1, one ADS solenoid to the "G" ADS SRV has failed in its normal position.

What SRV mode(s) of operation will function or not function when required to depressurize the reactor?

A. The ADS mode of operation will not function.
The RELIEF mode of operation will not function.

B. The ADS mode of operation will function.
The RELIEF mode of operation will function.

C. The ADS mode of operation will not function.
The RELIEF mode of operation will function.

D. The ADS mode of operation will function.
The RELIEF mode of operation will not function.

Proposed Answer: B

Explanation (Optional):

SRV "G" is an ADS valve with 3 solenoids, of which 2 are ADS. A loss of 1 ADS solenoid will not affect its ability to operate in the ADS or manual mode.

A. Incorrect – will still operate in both modes due to backup solenoid valve.
B. Correct
C. Incorrect - will still operate in both modes due to backup solenoid valve.
D. Incorrect - will still operate in both modes due to backup solenoid valve.

Per TM-OP-83E, Rev.0

In the event of an ADS initiation signal, either or both of the solenoids will reposition, closing off the vent path, and directing 150 psig nitrogen to the pneumatic cylinder to open the valve.

Technical Reference(s): TM-OP-083E-ST, Rev.0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 A2.01	
	Importance Rating	3.3	

Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of main steam flow inputs

Proposed Question: Common 17

The plant is operating at 100% power with the Feed Water Level Control System (FWLC) in its normal alignment.

A steam flow input to the FWLC system fails such that reactor water level decreases and stabilizes at a lower reactor water level (+25").

All other plant systems are operating properly

Which ONE of the following describes the malfunction that occurred to the steam flow input to the FWLC system and what operator actions are required IAW ON-145-001 "RPV Level Control Malfunction"?

A. The steam flow input to the FWLC system has failed high.
Transfer FWLC to "SEL" Level Selected.

B. The steam flow input to the FWLC system has failed low.
Transfer FWLC to "SEL" Level Selected.

C. The steam flow input to the FWLC system has failed high.
Transfer FWLC from Auto to Manual and depress "1 ELEM" control.

D. The steam flow input to the FWLC system has failed low.
Transfer FWLC from Auto to Manual and depress "1 ELEM" control.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – the steam flow input has failed low. A manual transfer to 1 element is required by procedure.
B. Incorrect - A manual transfer to 1 element is required by procedure.
C. Incorrect - the steam flow input has failed low.
D. Correct

ON-145-001 step 2.3 describes input failure

Per ON-145-001, Rev.20

3.6 If a loss of steam flow or Feed flow signal occurred, Place in Single Element Control as follows:

3.6.1 Verify FW LEVEL CTL]DEMAND SIGNAL LIC-C32-1R600 controller responding correctly and maintaining level ≤54" and ≥13".

3.6.2 Place FW LEVEL CTL/DEMAND SIGNAL LIC-C32-1 R600 controller in MANUAL.

3.6.3 IF conditions as described in CAUTION above exist, Adjust, as necessary, FW LEVEL CTLJDEMAND SIGNAL LIC-C32-1 P600 to raise and maintain RPV water level >30" as indicated on the operable level indicators LIC-C32-1 R606A(B)(C).

3.6.4 Depress Green 1 ELEM pushbutton for 1 OR 3 ELEMENT LEVEL CONTROL HS-106102.

Technical Reference(s): ON-145-001, Rev.20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	261000 K1.09	
Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Primary Containment Isolation System

Proposed Question: Common 18

The following conditions exist:

- Unit 1 is in Mode 2; Heatup in progress, with IRMs on Range 8.
- Unit 2 is at 100 percent power with all equipment OPERABLE.
- Unit 2 RPV water level dropped to -45 inches, and is now at +18 inches and stable following the event.

Which of the following describes the response of Reactor Building Ventilation System Zones (I, II, II) and the Standby Gas Treatment (SGTS) System?

	Zone I	**Zone II**	**Zone III**	**SGTS Fan "A"**	**SGTS Fan "B"**
A.	Does **NOT** Isolate	Isolates	Isolates	Starts	Starts
B.	Does **NOT** Isolate	Isolates	Does **NOT** Isolate	Starts	Remains Off
C.	Isolates	Does **NOT** Isolate	Isolates	Starts	Starts
D.	Isolates	Does **NOT** Isolate	Does **NOT** Isolate	Remains Off	Starts

Proposed Answer: A

Explanation (Optional):

K/A Match Justification:
High drywell pressure is an EOP entry condition and triggers the system responses.

A is correct.
B is incorrect. Zone III isolates, and SGT Fan "B" starts.
C is incorrect. Zone II isolates, and Zone I does NOT isolate.
D is incorrect. Zones II and III isolate, and Zone I does NOT isolate. SGT Fan "A" starts.

Technical Reference(s): TM-OP-70-ST, Rev.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP070/1954
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 2.4.2	
	Importance Rating	3.9	

Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (Standby Gas Treatment System)

Proposed Question: Common 19

A transient has occurred on Unit 1 with the following conditions.

- RPV Level is 30" and steady
- RPV Pressure is 800 psig and slowly lowering
- Drywell Pressure is 1.5 psig and rising
- Drywell Temperature 145 degrees F. and steady
- Suppression Pool Temperature is 85 degrees F. and steady
- The RB SPING release rate exceeds the HI-HI alarm

Which ONE of the following EOPs must be entered AND what is the required status of the Standby Gas Treatment System (SGTS)?

A. EO-100-103 "PC Control" and EO-100-104 "Secondary Containment Control" must be entered. NO other EOP entries are required.
SGTS is NOT required to be running.

B. EO-100-102 "RPV Control" must be entered. No other EOP entries are required.
SGTS is NOT required to be running.

C. EO-100-102 "RPV Control" and EO-100-103 "Primary Containment Control" must be entered. NO other EOP entries are required.
SGTS is required to be running.

D. EO-100-104 "Secondary Containment Control" must be entered. NO other EOP entries are required.
SGTS is required to be running.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - SGTS is required per EO-100-104 due to SPING alarm. No entry conditions exist for EO-100-103
B. Incorrect - SGTS is required per EO-100-104 due to SPING alarm. No entry conditions exist for EO-100-102
C. Incorrect - No entry conditions exist for EO-100-102 or EO-100-103.
D. Correct – SPING Alarm requires entry to EO-100-104 and SGTS initiation.

Technical Reference(s): EO-000-104 bases document (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K4.02	
	Importance Rating	2.9	

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Circuit breaker automatic trips

Proposed Question: Common 20

Unit 1 is in MODE 1. The ESS Bus 1B normal supply breaker tripped, and its alternate supply breaker failed to automatically close. DG "B" automatically started, and tied to ESS Bus 1B supplying all loads on the bus.

Which one of the following conditions will directly cause the DG supply breaker to ESS Bus 1B to automatically trip?

A. Generator High Differential Overcurrent.

B. Generator Overvoltage.

C. ESS Transformer T-211 lockout relay trips.

D. Unit 2 Core Spray Division 2 LOCA logic actuates.

Proposed Answer: A

Explanation (Optional):

A. Correct. Because EDG is running in Emergency
B. Incorrect - This does not generate an engine shutdown signal, and is therefore not going to trip the output breaker.
C. Incorrect - No impact, as the "normal" ESS transformer is not tied to the bus.
D. Incorrect - This would trip it only if the EDG were in parallel with the Offsite source at this time.

Technical Reference(s): TM-OP-024-ST, Rev.8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # TMOP004/10121

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215002 A4.06	
	Importance Rating	2.7	

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

Proposed Question: Common 21

Unit 1 is operating at 70% during power ascension. The SO-178-004 APRM Gain Adjustment surveillance test is in progress. APRM 1 has been bypassed per OP-178-002 Section 2.2.

When APRM 1 is returned to service (Unbypassed), what is the response of the Rod Block Monitor (RBM)?

A. RBM “A” reference APRM swaps from APRM 3 back to APRM 1. RBM “A” initiates a null sequence.

B. RBM “A” reference APRM swaps from APRM 4 back to APRM 1. RBM “A” does NOT initiate a null sequence.

C. RBM “B” reference APRM swaps from APRM 2 back to APRM 1. RBM “B” initiates a null sequence.

D. RBM “B” reference APRM swaps from APRM 4 back to APRM 1. RBM “B” does NOT initiate a null sequence.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – the first alternate is APRM 3 not APRM 4 and a null sequence is initiated.
C. Incorrect. – not the RBM “B”
D. Incorrect - not the RBM “B”

Technical Reference(s): TM-OP-78K-ST, Rev.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K3.03	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)

Proposed Question: Common 22

A fault occurs in 250 VDC Switchgear 1D662, causing both the battery charger output breaker to trip and the fuse from the battery to blow.

Which of the following loads would be affected by this event?

A. HPCI Auxiliary Oil Pump, 1P213.

B. Control Room Annunciators, 1C651.

C. RCIC Pump Disch Valve, HV-149F013.

D. “D” EDG Auto Start Solenoids, XV003445-D1/D2.

Proposed Answer: A

Explanation (Optional):

A. Correct – HPCI Aux Oil Pump is powered from Div 2 MCC supplied by 1D662.

B. Incorrect – Not powered from Div 2 250VDC; 1C651 Control Room Annunciators are powered from 1D635.

C. Incorrect – RCIC Discharge valve is powered from Div 1 MCC supplied by 1D652.

D. Incorrect – Not powered from Div 2 250VDC; D EDG Start solenoids are powered from 1D644.

TM-OP-088 - page 14
Loss of 250 VDC power can have a major impact on plant operation. It does have varying degrees of significance, from loss of DC lube oil pumps to loss of power to motor operated HPCI and RCIC Valves.

Loss of 250 VDC power to plant systems may result in system trouble alarms and other indications of equipment malfunction. Operators respond by making manual switching operations to restore DC power to the 250 VDC buses and to connected loads. For detailed information, refer to ON-188-001, Loss of 250 VDC Bus.

Technical Reference(s): TM-OP-088, Rev.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP088/1383/1
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000 A3.01	
Importance Rating	3.2	

Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights

Proposed Question: Common 23

The NPO reports the following indications on 1D652, 250V DC Load Center ground detection system.

- Both Positive (+) and Negative (-) lights are dimmer than normal
- The Positive (+) light is much brighter than the Negative (-) light.

WHICH ONE of the following is the status of grounds on the 250V DC bus?

A. A ground exists ONLY on the Positive (+) bus.

B. A ground exists ONLY on the Negative (-) bus.

C. Grounds exist on both buses with the ground of the greater magnitude on the Positive (+) bus.

D. Grounds exist on both buses with the ground of the greater magnitude on the Negative (-) bus.

Proposed Answer: D

Explanation (Optional):

Per TM-OP-088-ST, Rev.2, page 14.

Ground Detection

The 250 VDC System is an ungrounded system. The detection of a ground on the positive or negative side presents no immediate danger or loss of power generation; however, it should be immediately corrected before a ground occurs on the opposite side of the system causing a system short through ground.

Two sets of lights are used for the ground detection system and when no grounds are present, both lights are dimly lit. If one light burns bright and the other light is dim or out, then a single ground has occurred on the side with the dimly lit light. If grounds are present on both buses then the side with the dimly lit light has the grounds of greater magnitude. For detailed information of procedures for ground detection, refer to OP-188-001 (Unit 1) or OP-288-001 (Unit 2), 250 VDC System.

A. Incorrect. Grounds exist on both busses.
B. Incorrect. Grounds exist on both busses.
C. Incorrect. Bus with the dimmer light has the ground of greater magnitude.
D. Correct.

Technical Reference(s): TM-OP-088-ST, Rev.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP088/10131

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

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: Common 24

It is required to synchronize DG A to the grid to restore offsite power to 4.16KV bus ESS 1A201.

Which ONE of the following describes what the operator must ensure during this evolution IAW OP-024-001 "Diesel Generators"?

A. Synchroscope XI-00037 is rotating in the slow (counterclockwise) direction.
That running volts are slightly more than incoming volts on red scale 4KV Diff AC Volts XI-00036.

B. Synchroscope XI-00037 is rotating in the slow (counterclockwise) direction.
That running and incoming volts are matched on red scale 4KV Diff AC Volts XI-00036.

C. Synchroscope XI-00037 is rotating in the fast (clockwise) direction.
That running volts are slightly more than incoming volts on red scale 4KV Diff AC Volts XI-00036.

D. Synchroscope XI-00037 is rotating in the fast (clockwise) direction.
Running and Incoming volts are matched on red scale 4KV Diff AC Volts XI-00036.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – voltages must be matched per the procedure.
B. Correct
C. Incorrect – Synchroscope must be rotating in the slow direction and voltages must be matched by procedure.
D. Incorrect - Synchroscope must be rotating in the slow direction by procedure.

Note: Plant conditions above are for synchronizing back to offsite power which is infrequent and more complex than normal diesel test operation.

Per OP-024-001, Rev.49, step 2.9.3

Synchronize Diesel Generator A(B)(C)(D)(E), supplying 4.16KV ESS Bus 1A(1B)(1C)(1D), to preferred source of offsite power as follows:

a. Comply with TS 3.8.1 as applicable.

b. At Diesel Engine Control Panel 0C521A(B)(C)(D)(E), Place DG A(B)(C)(D)(E) Synchronization Auto Control Select switch 43SYN (Synchronizing Override (Emerg) for DG E) to Bypass (Override for DG E) to transfer control to Plant Operating Benchboard 0C653.

c. On Plant Operating Benchboard 0C653, Parallel diesel generator and offsite power as follows:

(1) Place Xfmr 101(211)(111)(201) – Bus 1A(1B)(1C)(1D) Sync Sel switch to ON.

(2) Adjust DG A(B)(C)(D) Voltage Adjust HS-00053A(B)(C)(D) and Speed Governor HS-00054A(B)(C)(D) to achieve following:

(a) **Running and Incoming volts matched on red scale 4KV Diff AC Volts XI-00036.**
(b) **Synchroscope XI-00037 rotating in SLOW (counterclockwise) direction.**

(3) WHEN Synchroscope XI-00037 at or slightly before "12 o'clock" position, Close Xfmr 101(211)(111)(201) to Bus 1A(1B)(1C)(1D) Bkr 1A20101(1A20209)(1A20301)(1A20409).

Technical Reference(s): OP-024-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.02	
	Importance Rating	3.0	

Knowledge of INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Cross-over to other air systems

Proposed Question: Common 25

Unit One is operating at rated power when an air leak occurs on the common header downstream of the instrument air (I/A) receivers. The 1A I/A compressor is in LEAD and the 1B I/A compressor is in STANDBY.

- Prior to the leak, pressure was 101 psig being maintained by 1A I/A compressor
- Air pressure decreased to 80 psig over twenty minutes.
- Air pressure then recovered to 90 psig.

Which one of the following statements describes the expected status of the instrument air system at this time?

A. Both instrument air compressors are running fully loaded.
The Service Air cross-tie PCV-12560 is controlling pressure.

B. Both instrument air compressors are running fully loaded.
The Unit 1 to Unit 2 IA cross-tie valve is open supplying the header.

C. The 1A instrument air compressor is running fully loaded.
The 1B instrument air compressor is running at 50% load.
The Service Air cross-tie PCV-12560 is open.

D. The 1A instrument air compressor is running fully loaded.
The 1B instrument air compressor is running at 50% load.
The Unit 1 to Unit 2 IA cross-tie valve is supplying the header.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect - The SA cross-tie valve is open at 90psig
C. Incorrect – The standby compressor runs fully loaded until header pressure reaches 96 psig.
D. Incorrect - The standby compressor runs fully loaded until header pressure reaches 96 psig. The SA cross-tie regulator is open.

Per TM-OP-018-ST, Rev.3

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SERVICE AIR
The Service Air Compressors serve as backup to the Instrument Air Compressors. The manual isolation valves between the Service Air System and the Instrument Air System are normally open. The Service Air Crosstie to Instrument Air (PCV-12560) senses pressure at the outlet of the compressors. It starts to open at 95 psig, and comes fully open at 90.

Page 10

The Standby Compressor, with its local control switch in the AUTO position, and Load Limit Select Switch in FULL, will automatically start when system pressure drops to 87 psig, and will run (at 100 percent load) until system pressure is restored above 96 psig, at which time it will shift to 50 percent load. If system pressure continues to rise, the compressor will completely unload at 102 psig. If system pressure remains above 102 psig for 30 minutes, the Standby Compressor will shut down.

Technical Reference(s): TM-OP-018-ST, Rev.3 (Attach if not previously provided)
ON-118-001, Section 2.0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank #
Modified Bank # TMOP018/1769 (Note changes or attach parent)
New

Question History: Last NRC Exam

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

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

Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Proposed Question: Common 26

Unit one is operating at 75 percent power when a leak develops on RBCCW suction piping. Before any actions can be taken, the following parameters are noted:

- Head tank is empty.
- System pressure is 55 psig.

Which ONE of the following statements describes the status of the RBCCW pumps and what action(s) is (are) required to prevent damage to the pump(s)?

A. ONLY one RBCCW pump is running.
Depress and Hold the RBCCW Pump STOP push button and open the respective MCC breaker locally.

B. ONLY one RBCCW pump is running.
Depress the RBCCW Pump STOP push button.

C. BOTH RBCCW pumps are running.
Depress and Hold the RBCCW Pumps STOP push buttons and open their respective MCC breakers locally.

D. BOTH RBCCW pumps are running.
Depress the RBCCW Pumps STOP push buttons.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. – Both RBCCW pumps will be running.

B. Incorrect – Both RBCCW pumps will be running. MCC breaker must be opened locally while holding depressed the RBCCW pump pushbuttons.

C. Correct

D. Incorrect - MCC breaker must be opened locally while holding depressed the RBCCW pump pushbuttons.

Per TM –OP-014, Rev. 1

Technical Reference(s): TM-OP-014-ST, Rev.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 # TMOP014/1694/1 (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

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

Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following: Control rod drive mechanisms

Proposed Question: Common 27

The CRD Hydraulic System is operating normally at rated power.

The power supply to the CRD flow controller fails.

Which ONE of the following describes the impact of the failure on the CRD system?

A. The Flow Control Valve fully opens.
CRD pump trips on high flow

B. The Flow Control Valve closes.
CRDM temperature will increase.

C. The Flow Control Valve fully opens.
CRD charging header pressure decreases

D. The Flow Control Valve closes.
CRDM drive header pressure decreases.

Proposed Answer: B

Explanation (Optional):

NOTE: This event occurred within the past 2 years at the plant.

A. Incorrect – the FCV closes
B. Correct
C. Incorrect – the FCV closes
D. Incorrect – drive header pressure is maintained because the FCV does not FULLY close.

Technical Reference(s): ON-155-007, Rev.19 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP055/2419
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215001 A1.02	
	Importance Rating	2.5	

Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Detector Position

Proposed Question: Common 28

Unit 1 is operating at 100 percent power. Traversing Incore Probe (TIP) Machine "A" is preparing to run a TIP trace. The following conditions exist:

- ALL other TIPs are in their shield chambers.
- TIP Machine "A" is operating in the AUTO Mode.
- The TIP detector has left the indexer and is passing the Bottom Core Limit.
- A feedwater level control transient caused RPV water level to lower to minus (-) 5 inches and stabilize at +19 inches.

Assuming no Operator actions, which one of the following are the automatic actions of the "A" TIP Machine?

The "A" TIP...

A. continues to the Top Core Limit position and stops.

B. immediately stops movement and shifts to manual operation.

C. immediately reverses, withdraws from the core into the shield chamber.

D. continues to the Top Core Limit, reverses and withdraws into the shield chamber.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - The TIP will withdraw and isolate since RPV level lowered to +13 inches. (Note: Alternate choice if the candidate believes the isolation signal is -38 inches, the setpoint for the TIP Purge valve. Without an isolation signal, a TIP in Auto would progress to the top of the core and stop.)

B. Incorrect – The TIP does not shift to manual in this condition.

C. Correct – Since RPV level lowered to the +13" setpoint, the TIP will automatically reverse and withdraw to the shield chamber (and the ball valve closes on limit switch interlock to isolate).

D. Incorrect – TIP immediately reverses and automatically retracts.

Technical Reference(s): OP-178-001, Rev. 17 (Attach if not previously provided)
TM-OP-078F, Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # TMOP078F/2316/1 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 K6.05	
	Importance Rating	2.7	

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

Proposed Question: Common 29

During a startup on Unit 2, with power at 20%, the total Steam flow input to the RWM fails downscale.

How will this failure affect operation of the RWM?

A. Rod blocks will be enforced because the system senses reactor power below the Low Power Alarm Point.

B. Rod blocks will be enforced because the system senses reactor power below the Low Power Setpoint.

C. Rod blocks will NOT be enforced because the system senses reactor power below the Low Power Alarm Point.

D. Rod blocks will NOT be enforced because the system senses reactor power below the Low Power Setpoint.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – being below the LPAP does not enforce blocks, being below the LPSP enforces blocks.
B. Correct - With the total steamflow signal failed low, the RWM system will sense low power (below LPSP) and enforce rod blocks despite actual power at 25%
C. Incorrect – rod blocks will be enforced.
D. Incorrect – rod blocks will be enforced.

Per TM-OP-031D-ST, Rev.0,

Page 3

As reactor power level is raised, the effect that movement of a single rod can have on the overall flux distribution becomes relatively smaller and smaller. Therefore, maintaining rod worths low is critical only when operating at relatively low power levels. For this reason, the RWM enforces adherence to a withdrawal or insertion sequence only when core power is below and adjustable Low Power Setpoint (LPSP). The setpoint is currently established about 10 percent of rated thermal power (RTP) as sensed by total steam flow from the Feedwater Level Control (FWLC) system.

Page 4

At power levels above the LPSP, the system does not impose rod blocks as a result of out of sequence movements. This is due to the fact that individual rod worths above this power level are of such a magnitude that they will not pose any risk of fuel damage from a rod drop accident.
In order to allow the operator to verify and correct control rod alignment during a shutdown, the RWM begins to track control rod movement at approximately 20 percent rated thermal power. This is known as the Low Power Alarm Point (LPAP). Between the LPAP and the LPSP, the area known as the Transition Zone, the RWM recognizes and displays any out of sequence rod positions, but does not generate any rod blocks due to those out of sequence rod positions. It displays the out of sequence rod positions, which allows the operator to correct them prior to entering a condition that could cause restricted rod movement. Above the LPAP the RWM is auto bypassed and does not monitor the control rod movements.

Technical Reference(s): TM-OP-031D-ST (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 A1.04	
	Importance Rating	3.3	

Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: reactor water level

Proposed Question: Common 30

During a Reactor Recirculation Pump start, which ONE of the following describes the response of indicated reactor vessel water level and the reason for that response?

A. Reactor water level increases due to a displacement of water from the downcomer into the core shroud area.

B. Reactor water level increases due to the displacement of water from the core shroud area into the downcomer region.

C. Reactor water level decreases due to the displacement of water from the downcomer into the core shroud area.

D. Reactor water level decreases due to the displacement of water from core shroud area into the downcomer region.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – indicated level decreases
B. Incorrect – indicated level decreases
C. Correct. - Reactor water level decreases due to the displacement of water from the downcomer into the core shroud area as the Recirculation Pump discharge pressure rises.
D. Incorrect – the increase is due to displacement of water from the downcomer into the core shroud area.

Note before step 2.3.28 on page 18 of OP-164-001, rev 47. discusses level decrease on start.

Technical Reference(s): OP-164-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:	Bank #	TMOP064C/2526	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	219000 K4.08	
	Importance Rating	2.9	

Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE design feature(s) and/or interlocks which provide for the following: Adequate pump net positive suction head

Proposed Question: Common 31

Following a LOCA and a failure to scram (ATWS) on Unit 1, all control rods were manually inserted and a Rapid Depressurization was performed. The following conditions now exist:

- Reactor water level is -75 inches, rising slowly.
- "A" Core Spray loop is injecting at 6,200 gpm.
- "B" Core Spray loop is available.
- "A" RHR loop flow is maximized in Supp Pool Cooling.
- "B" RHR loop is unavailable due to a suction pipe leak.
- Supp pool level is 18 feet, lowering slowly.
- Supp pool temp is 185 degrees F.

Which ONE of the following is correct regarding ECCS pump operation as Suppression Pool level continues to lower and reaches 17 feet?

A. Throttle or stop "A" Core Spray flow to lower <6,000 gpm AND Maintain "A" RHR loop flow maximized.

B. Throttle or stop "A" Core Spray flow to lower <6,000 gpm AND Throttle or stop "A" RHR loop flow.

C. Maintain "A" Core Spray flow at 6,200 gpm to restore RPV level AND Throttle or stop "A" RHR loop flow.

D. Maintain "A" Core Spray flow at 6200 gpm to restore RPV level AND Maintain "A" RHR loop flow maximized.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – RHR loop flows must be throttled or stopped since below the curve. Current plant conditions do not authorize operation irrespective of these limits.

B. Correct – CS loop flow is below the curve and RHR flow is below the curve. Current plant conditions do not authorize operation irrespective of these limits.

C. Incorrect – CS loop flows must be throttled or stopped since below the curve. (Note: alternate choice if candidate believes lower curve is for RHR not CS.)

D. Incorrect – CS and RHR loop flows must be throttled or stopped since below the curve. (Note: alternate choice if candidate believes plant conditions require operation beyond pump limits.)

NOTE: RHR maximized is defined to be as close to 10,000 gpm flow as possible.

Per EO-000-100 Caution VL, Page 8

Caution VL OPERATION OF RHR
OR CORE SPRAY WITH SUCTION FROM SUPP POOL
AND PUMP FLOW BELOW VL
MAY RESULT IN EQUIPMENT DAMAGE.

EO Basis:
Vortex limits are the lowest suppression pool water levels above which air entrainment is not expected to occur in pumps taking suction on the pool. These levels are functions of ECCS flow. Exceeding the limits can lead to air entrainment at the pump suction strainers.

NOTE: At SSES, RHR and Core Spray NPSH limits are included within and bounded by their respective vortex limits, since they are less restrictive than the vortex limits.

Per EO-000-103 PC Control, Page 13

Step SP/L-4 MAINTAIN SUPP POOL LEVEL ABOVE 17' USING ANY:

EO Basis:
Decreasing suppression pool water level has several negative effects:

1. Low level reduces the capacity of the primary containment to absorb all the energy from blowdown of the RPV and still remain below 65 psig.

2. Low level adversely affects the flow capacity of ECCS pumps taking suction from the suppression pool. Low level increases the probability of air entrainment and reduces NPSH.

TM-OP-049-ST, Rev. 03 page 9

The RHR suction strainers have a sufficient flow cross-section capacity to filter the insulation, paint chips, and other drywell debris that is assumed to be dislodged by LOCA jet forces and transported to the Suppression Pool through the downcomers along with corrosion products that exist in the Suppression Pool prior to a LOCA under worst case conditions, while maintaining strainer pressure drop below the maximum required to provide adequate NPSH and system flow.

Vortex limits are conservative in that they assume suction strainer blockage.

Technical Reference(s): EO-000-100 Caution VL (Attach if not previously provided)
EO-000-103 Step SP/L-4

Proposed references to be provided to applicants during examination: Fig 7 Vortex Limit Curve WITHOUT TEXT NOTES. (as provided on modified EOP flowchart)

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	226001 A3.03	
Importance Rating	2.8	

Ability to monitor automatic operations of the RHR/LPCI CONTAINMENT SPRAY SYSTEM MODE including: System flow

Proposed Question: Common 32

Given the following conditions:

- A LOCA has occurred on Unit 1
- Drywell Pressure is 18 psig and slowly lowering
- Supp Chamber and Drywell Sprays are in service with the "A" RHR Pump.
- RPV Level is -125 inches and slowly lowering
- RPV Pressure is 520 psig and slowly lowering

Which ONE of the following describes the status of Drywell Spray flow when RPV level reaches -129 inches and action required to remain in Drywell Spray mode?

A. Spray flow will rise since the HV-151-F048 HX Bypass valve will automatically open and 'C' RHR pump will start. Manually close the F048 valve to maintain spray flow < 10,000 gpm.

B. Spray flow will rise since the 'C' RHR pump will automatically start. Manually stop the 'C' RHR pump OR throttle sprays to maintain spray flow < 10,000 gpm.

C. Spray flow will lower since the HV-151-F017A will automatically open to align for injection. Manually close the F017A valve AND throttle spray flow to desired flow.

D. Spray flow will stop since the HV-151-F016A and F021A will automatically close. Place the S17A LOCA Isolation Manual Override in OVRD and reopen the spray valves to align sprays.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – On the LPCI initiation signal (-129"), the F048 receives an open signal. However, to align for sprays >1.72 psig, the operator previously placed the / LOCA Isolation Manual Override S17A keyswitch is in OVRD. The F048 remains closed.
B. Correct.
C. Incorrect – The F017 will not auto open until RPV pressure lowers to <420 psig.
D. Incorrect – On the LPCI initiation signal (-129"), the F016 and F021 receives a close signal. However, to align for sprays >1.72 psig, the operator previously placed the / LOCA Isolation Manual Override S17A keyswitch is in OVRD. The F016 and F021 remain open.

Per TM-OP-049-ST

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Valve F017 can be opened manually using the control switch when reactor pressure is less than 420 psig or when the associated F015 is closed (the normal lineup). F017 receives an open signal, preventing valve closure and opening the valve if closed that auto resets after 45 seconds on any of the following signals:

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The Drywell Spray Outboard Isolation Valve (HV-151-F016A and B) can be opened and closed from the associated 1C601 Panel (CLOSE-AUTO-OPEN) Control Switch (DRYWELL SPRAY OB ISO HV-151-F016A/B) with (amber- closed/red-open) valve position indication and are normally closed. The thermal overload device is normally defeated and placed in service for testing only. The F016 Valves are throttleable to control the flowrate into the drywell spray header to a maximum of 10,000 gpm.

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Both valves auto close on any of the following signals:

RPV Level Low <-129 inches

Drywell Pressure High >1.72 psig

Manual Initiation Div 1 or 2

With the auto closure signal present the closure signal can be overridden using the S17A/B, LOCA Isolation Manual Override Switch and then placing the valve control switch to OPEN. The override remains sealed in to the F021 and F016 Valves until the RHR Loop Initiation signal is reset or reset using the associated S17A/B Switch.

Technical Reference(s): TM-OP-049-ST, Rev. 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000 K1.08	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause- effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Reactor/turbine pressure control system: Plant-Specific

Proposed Question: Common 33

A plant startup is in progress on Unit 1.

- Main generator is tied to the grid.
- Reactor power is 22 percent.

The EHC piping downstream of the pumps develops a leak, and EHC FAS header pressure is steadily decreasing. Assuming **NO** operator action is taken, which of the following statements describes the expected plant response?

A. The Main Turbine trips; the Reactor scrams when Intercept Valves drift closed, resulting in a Generator Lockout after a time delay.

B. The Main Turbine trips; the Reactor scrams on high pressure when Bypass Valves drift closed after accumulators depressurize.

C. The Main Turbine Control and Stop Valves drift closed; the Reactor scrams when Control Valve ETS pressure is <500 psig.

D. The Main Turbine Control and Stop Valves drift closed; the Reactor scrams when Stop Valves reach 94.5 percent open.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Turbine trip will occur on low FAS pressure signal. Generator lockout will not scram reactor.

B. Correct – Below 30%, the scram on turbine trip is bypassed. Bypass valves will initially open to control pressure but have limited accumulator oil supply. The reactor will scram on high pressure, the backup scram for this event and condition.

C. Incorrect – Turbine trip will occur on low FAS pressure signal. Reactor Scram will not occur <30% power.

D. Incorrect – Turbine trip will occur on low FAS pressure signal. Reactor Scram will not occur <30% power.

Note: Valves drifting closed can occur if turbine trip function on low oil pressure does not occur.

Per TM-OP-093E-ST, Rev. 5 Page 9
Three Pressure Switches (PSL 10180A, B and C) sense FAS header pressure, and actuate on low pressure (1,100 psig). The switches input to a two-out-of-three logic scheme to generate a Main Turbine Trip signal and will activate Control Room annunciators

Per TM-OP-093E-ST, Rev. 5 Page 28
Additional accumulators serve the hydraulic supply lines to the actuators of the Bypass Valves. Three accumulators are attached to the FAS Line to the Control Pacs. These accumulators are pre-charged to 900 psig, and they will provide one minute of opening time for the bypass valves if a loss of EHC pressure and turbine trip occur.

Per ON-193-002, Rev. 14
For Main Turbine trip above 30% Reactor power, based on turbine first stage pressure, Stop Valve Closure or Control Valve fast closure causes Reactor Scram. Below this value, Scram is bypassed. Main Turbine Trip can be caused by Main Generator Trip on load reject or other electrical fault. These trips protect Main Generator from damage.

Technical Reference(s): TM-OP-093L-ST, Rev.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	AD045/1357/64	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000 A4.02	
	Importance Rating	2.8	

A4.02 – Ability to manually operate and/or monitor in the control room: System motor operated valves

Proposed Question: Common 34

Unit 1 is operating at rated power. The following occurred:

- HIGH COND DEMIN HEADER DIFF PRESS alarm is received.
- CONDENSATE FILTRATION SYSTEM HI-HI DP alarm is received.
- Condensate Demin and Condensate Filtration System differential pressures are 55 psid, slowly rising.

The plant responds as designed, what will be the FINAL position of the following valves? (assume no operator action)

	Cond Demin Bypass Valve HV-11621	CFS Bypass Valve FV-10572
A.	Closed	Closed
B.	Closed	Open
C.	Open	Open
D.	Open	Closed

Proposed Answer: B

Explanation (Optional):

A. Incorrect – FV-10572 opens
B. Correct
C. Incorrect – HV-11621 is closed.
D. Incorrect – FV-10572 opens.

Technical Reference(s): LA-11103-001, Rev.5 –A02 (Attach if not previously provided)
LA-1121-001, Rev.5-B01

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank # TMOP044/10723/1
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 X
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 A2.02	
	Importance Rating	3.1	

A2.02 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Feedwater heater isolation

Proposed Question: Common 35

Unit 2 is at 90% power when a High-High water level occurs on the 3A, 4A, and 5A Feedwater Heaters. A decrease in feedwater temperature has been noted by the operators.

Which ONE of the following describes the plant response and actions required to be taken IAW ON-247-001 "Loss of Feedwater heating Extraction Steam"?

A. The associated feedwater heater string will automatically isolate.
Reactor power must be immediately reduced to less than or equal to 85%.

B. The associated feedwater heater string will NOT automatically isolate.
Reactor power must be immediately reduced to less than or equal to 85%.

C. The associated feedwater heater string will automatically isolate.
Reactor power must be immediately reduced by at least by 25%.

D. The associated feedwater heater string will NOT automatically isolate.
Reactor power must be immediately reduced by at least by 25%.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – the string is manually removed from service
B. Correct
C. Incorrect - the string is manually removed from service. Reactor power must be reduced to less than or equal to 85%.
D. Incorrect - Reactor power must be reduced to less than or equal to 85%.

Per ON-247-001, Rev.21

Step 3.2

Reduce Reactor Power IAW RE Instructions in CRC Book to less than or equal to 85% RTP.

Step 3.8

If any feedwater heating lost and cannot be restored within 2 hours, Isolate affected feedwater string in accordance with OP-244-001 Condensate and Feedwater System.

Technical Reference(s): ON-247-001,Rev.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 ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 2.1.2	
	Importance Rating	3.0	

Conduct of Operations Knowledge of operator responsibilities during all modes of plant operation. (Offgas)

Proposed Question: Common 36

Unit 1 was at 90% power when an Offgas System automatic isolation occurred due to a valid Low Dilution Steam Flow signal.

Which ONE of following describe what the operators must ensure IAW ON-143-001 "MAIN CONDENSER VACUUM AND OFFGAS SYSTEM OFF-NORMAL OPERATION"?

A. Ensure Isolation occurred by checking SJAE Suct Iso Vlvs HV-10716, HV-10717, HV-10718 and HV-10719 are CLOSED, as indicated by Amber light ILLUMINATED at HS-10716.
Ensure valves HV-10701A SJAE Mn Stm Sup Iso, HV-10701B SJAE Mn Stm Sup Iso and HV-10107 Mn Stm SJAE Iso are OPEN.

B. Ensure Isolation occurred by checking SJAE Suct Iso Vlvs HV-10716, HV-10717, HV-10718 and HV-10719 are CLOSED, as indicated by Amber light EXTINGUISHED at HS-10716.
Ensure valves HV-10701A SJAE Mn Stm Sup Iso, HV-10701B SJAE Mn Stm Sup Iso and HV-10107 Mn Stm SJAE Iso are OPEN.

C. Ensure Isolation occurred by checking SJAE Suct Iso Vlvs HV-10716, HV-10717, HV-10718 and HV-10719 are CLOSED, as indicated by Amber light ILLUMINATED at HS-10716.
Ensure valves HV-10701A SJAE Mn Stm Sup Iso, HV-10701B SJAE Mn Stm Sup Iso and HV-10107 Mn Stm SJAE Iso are CLOSED.

D. Ensure Isolation occurred by checking SJAE Suct Iso Vlvs HV-10716, HV-10717, HV-10718 and HV-10719 are CLOSED, as indicated by Amber light EXTINGUISHED at HS-10716.
Ensure valves HV-10701A SJAE Mn Stm Sup Iso, HV-10701B SJAE Mn Stm Sup Iso and HV-10107 Mn Stm SJAE Iso are CLOSED.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – The amber light will be illuminated.
C. Incorrect- The main steam supply valves must be ensured OPEN.
D. Incorrect - The amber light must be illuminated. The main steam supply valves must be ensured OPEN

Per ON-143-001,Rev.20, step 3.5.7

IF Offgas System Automatic Isolation occurred, THEN Perform following:

a. Ensure Isolation occurred by checking SJAE Suct iso Vlvs HV-1 0716, HV-10717, HV-10718 and HV-10719 CLOSED, as indicated by Amber light ILLUMINATED at HS-10716.

b. IF isolation originated from Low Dilution Steam Flow, THEN Perform following:
(1) IF on Main Steam, THEN Ensure following valves are OPEN:
(a) HV-10701A SJAE Mn Stm Sup Iso.
(b) HV-10701 B SJAE Mn Stm Sup Iso.
(c) HV-10107 Mn Stm SJAE iso.

Technical Reference(s): ON-143-001,Rev.20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 1239, 1240, 10338 (As available)

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 X
55.43

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	223001 K5.01	
Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES : Vacuum breaker/relief operation

Proposed Question: Common 37

Which ONE of the following component failures, by itself, will result in Drywell pressure exceeding the containment internal design pressure limit?

A. Both SRV Vacuum Relief Valves on a tailpipe stuck open during a LOCA with ADS actuation.

B. Both SRV Vacuum Relief Valves on a tailpipe stuck closed during a LOCA with ADS actuation.

C. Both Suppression Pool to Drywell Vacuum Breakers on a downcomer failed open during a design basis LOCA.

D. Both Suppression Pool to Drywell Vacuum Breakers on a downcomer failed closed during a design basis LOCA.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - This failure is not a containment internal design pressure concern since these VRVs discharge into the drywell (bounded by a MSL rupture). VRVs discharge to drywell, there is a common misconception that VRVs discharge to the SC airspace.

B. Incorrect - This failure is not a containment internal design pressure concern but instead a tailpipe loading concern. SRV tailpipe rupture in Supp Chamber airspace is a Pressure Suppression Limit concern.

C. Correct – Failed open vacuum breakers under these conditions would bypass suppression function and exceed containment design pressure.

D. Incorrect – Failed closed vacuum breakers under these conditions is not a containment internal design pressure concern per TS Bases.

TS Bases 3.6.1.6

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Suppression chamber-to-drywell vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber floor.

The safety analyses assume that the vacuum breakers are closed initially and are open at a differential pressure of $\leq$ 2.81 psid (Ref. 1). Additionally, one of the five vacuum breaker pairs is assumed to fail in a closed position (Ref. 1). <u>The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. Design Basis Accident (DBA) analyses require the vacuum breakers to be closed initially and to remain closed and leak tight, with the suppression pool at a positive pressure relative to the drywell.</u>

Technical Reference(s): TS Bases 3.6.1.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP059/10358/1

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000 K4.02	
	Importance Rating	3.3	

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

Proposed Question: Common 38

The Halon System in the Upper Relay Room has the following conditions.

- One Product of Combustion (Smoke) Detector has tripped.
- Thermal (Heat) Detectors in the same room are not tripped.
- A control room Simplex Panel alarm was received.

Which ONE of the following describes the actuation of the Upper Relay Room Halon system and why?

A. No Halon discharge will occur because two Combustion (Smoke) detectors are required to actuate a Halon discharge.

B. No Halon discharge will occur because the Combustion (Smoke) Detector is a pre-alarm. A single Thermal (Heat) detector is required to actuate a Halon discharge.

C. No Halon discharge will occur because the Combustion (Smoke) Detector is a pre-alarm. Two Thermal (Heat) detectors are required to actuate a Halon discharge.

D. A Halon discharge will occur because a single detector of either type will initiate a Halon discharge.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – No discharge will occur, POC is a pre-alarm
B. Correct - The POC Detector is a pre-alarm and with that, only one Thermal Detector is needed to initiate Halon discharge.
C. Incorrect – only one thermal must actuate.
D. Incorrect – No discharge will occur

Technical Reference(s): TM-OP-13,Rev.5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP013/2295

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK2.01	
	Importance Rating	3.6	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Recirculation system.

Proposed Question: Common 39

Unit 1 is operating at 85 percent power with both Recirc Pumps at 86 percent speed. The "A" Recirc Pump Scoop Tube locks when a fuse blows in the control circuit.

SELECT the expected response of the Recirculation Pumps if Condensate Pump B trips, level decreases to 28 inches, and then is restored to +35 inches by feedwater.

A. Pump A tripped - Pump B speed is 30 percent.

B. Pump A tripped - Pump B speed is 48 percent.

C. Pump A speed is 86 percent - Pump B speed is 30 percent.

D. Pump A speed is 86 percent - Pump B speed is 48 percent.

Proposed Answer: D

Explanation (Optional):

A	Incorrect - The locked pump remains at 86 percent.
B	Incorrect - The locked pump remains at 86 percent.
C	Incorrect - The level drop below 30 inches in conjunction with the cond pump pressure < 100 delivers a 45 percent runback signal.
D	Correct.

Per ON-164-002, Section 4.3.11.b

b. Limiter #2 (48%) runback initiated by:
(1) Any Circulating Water Pump protective trip.
(2) RPV low water level (+ 30") and any of following:
(a) Feedwater flow A, B, or C decrease to
□ 20%.
(b) Any Condensate Pump discharge pressure
⌝ 100 psig.
(c) Auto isolation of Feedwater Heaters String
A, B or C due to high level in Feedwater
Heaters 1 or 2.

Technical Reference(s): ON-164-002 Rev.25 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP064A/2581
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.03	
	Importance Rating	3.2	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Battery status: Plant-Specific

Proposed Question: Common 40

Which ONE of the following describes how a loss of Division 1 AC power would affect the 24 VDC system on Unit 1?

A loss of AC would affect_______.

A. two battery chargers supplied from Essential Instrument AC Panels providing power to one distribution panel and one battery bank.

B. two battery chargers supplied from Non-essential 480 VAC MCC's providing power to one distribution panel and one battery bank.

C. one battery charger supplied from essential instrument AC Panels providing power to one distribution panel and two battery banks.

D. one distribution panel and two battery banks supplied from Non-essential 480 VAC MCC.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – 24 VDC powered from essential
C. Incorrect – two chargers not one.
D. Incorrect – 24 VDC powered from essential

Technical Reference(s): TM-OP-75-ST, Rev.75 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________________ (As available)

Question Source: Bank # TMOP075/10102/

Modified Bank # ______________ (Note changes or attach parent)

New ______________

Question History: Last NRC Exam ______________

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:

		RO	SRO
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK1.03	
	Importance Rating	2.9	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Electrical bus divisional separation

Proposed Question: Common 41

Both Units are at 100% power.

Which ONE of the following describes the plant response to a loss of 125 VDC bus 1D622?

A. D/G B and D are inoperable for auto start.

B. Lockout protection has been disabled for Startup Bus 10 and Startup Transformer 10.

C. HPCI Steam Leak Detection Isolation is inoperable.

D. Reactor Recirc Pumps 1A trips.

Proposed Answer: D

Explanation (Optional):

A	Incorrect – Only Diesel Generator B has lost all control power and will not start.
B	Incorrect - Startup Bus 20 and Startup Transformer 20 has lost all lockout protection.
C	Incorrect - HPCI steam leak detection is operable. HPCI Manual and Automatic Initiation are inoperable. (1D624-01) & (1D624-06)
D	Correct -

Per ON-102-620, Rev.6, page 8

IMPACT STATEMENTS FOR LOSS OF 1D624

Technical Reference(s): ON-102-620,Rev.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # AD045/1357

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA1.05	
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : Reactor /turbine pressure regulating system.

Proposed Question: Common 42

Unit 1 is at 100 percent power.

Which one of the following describes the result if the in-service EHC pressure regulator's output fails high?

A. Backup regulator controls reactor pressure at a higher value.

B. Max Combined Flow limit maintains total steam flow at 100 percent.

C. Reactor scram on MSIV closure.

D. Turbine trip caused by TCVs failing closed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Backup regulator will not take control when the in service regulator fails high.
B. Incorrect - Max combined flow limit is set for 125% and MSIV closure cannot be avoided.
C. Correct
D. Incorrect - TCVs will be at 100 percent open. Note: Plausible since Turbine would trip on generator lockout, reverse power if candidate believes valves fail closed.

Technical Reference(s): ON-193-001, Rev.12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank # TMOP093L/1638

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ________

55.43 ________

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295006 2.4.49	
Importance Rating	4.0	

Emergency Procedures / Plan Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (Scram)

Proposed Question: Common 43

With Unit 1 operating at rated power, which ONE of the following plant conditions below requires an Immediate Reactor Scram?

A. Feedwater transient with RPV level lowering to 19 inches, then begins to rise.

B. EHC malfunction with RPV pressure rising to 1090 psig, then begins to lower.

C. Recirc Pump trip that results in entry into Region II of the Power/ Flow Map.

D. CRD hydraulic transient with two control rods drifting into the full in position.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Scram not required. Feedwater transients that lower to the scram setpoint require a scram. The scram setpoint is ≥13 inches however, the ON-145-001 immediate operator action is to scram when RPV level lowers to 18 inches.

B. Correct – Scram required. The EHC malfunction results in reactor pressure above the scram setpoint. If during a plant transient the operator observes an existing situation where the Reactor Protection System did not initiate an automatic reactor scram when required, he/she shall initiate a manual reactor scram IAW OP-AD-002

C. Incorrect – Scram not required. A Recirc pump trip can result in entry into restricted regions of the power to flow map. Region II requires immediate exit but not immediate scram IAW ON-178-002. Region I may require an immediate scram.

D. Incorrect – Scram not required. ON-155-001, Control Rod Problems directs an immediate operator scram if 3 or more control rods drift given this plant condition.

OP-AD-001, Rev. 38 page 17

6.2 FAILURE OF THE REACTOR PROTECTION SYSTEM TO PERFORM ITS FUNCTION

6.2.1 If during a plant transient the operator observes an existing situation where the Reactor Protection System did not initiate an automatic reactor scram when required, he shall initiate a manual reactor scram. An Existing Scram Condition is listed as requiring immediate operator action.

Technical Reference(s): OP-AD-001, Rev. 38 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 2.4.50	
	Importance Rating	3.3	

Emergency Procedures / Plan Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (Control Room Abandonment)

Proposed Question: Common 44

The control room has been abandoned and ON-100-009 “Control Room Evacuation” has been implemented.

Which ONE of the following describes indications available and systems operation when control is transferred to the Remote Shutdown Panel?

A. RCIC will NOT shutdown when RPV level reaches 54 inches and must be manually tripped by the operator.
Open indication for PSV-141-F013A SAFETY RELIEF VALVE “A” can only be verified by monitoring reactor pressure.

B. RCIC will NOT trip on overspeed and must be manually tripped by the operator the operator.
Open indication for PSV-141-F013A SAFETY RELIEF VALVE “A” can only be verified by monitoring reactor pressure.

C. RCIC will NOT shutdown when RPV level reaches 54 inches and must be manually tripped by the operator.
Open indication for PSV-141-F013A SAFETY RELIEF VALVE “A” can only be verified by the OPEN solenoid energized light.

D. RCIC will NOT trip on overspeed and must be manually tripped by the operator the operator.
Open indication for PSV-141-F013A SAFETY RELIEF VALVE “A” can only be verified by the OPEN solenoid energized light.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – RCIC WILL trip on overspeed
C. Incorrect – No SRV Open indication is available at the RSP
D. Incorrect – RCIC WILL trip on overspeed. No SRV Open indication is available at the RSP.

.

Per ON-100-009, Rev.14

Wide range level instrumentation is increasingly less accurate as pressure decreases. At approximately 200 psig, wide range level instrument indicates +60" when actual level is 0". Temperature indication is not available at the shutdown panel to monitor reactor coolant temperature.

Page 11, step 4.6.4

CAUTION - RC1C will not trip on high vessel level +54"

Page 25 of 64, Rev. 14.

9. HSS-1 5111 B CONTROL TRANSFER SWITCH R
b. PSV-141-FO13A SAFETY RELIEF VALVE A
(1) CONTROL (NO INDICATION ON RSP)
(2) DEFEAT RELIEF FUNCTION

Technical Reference(s): ON-100-109, Rev.14 (Attach if not previously provided)
OP-150-001, Rev.27

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 2.4.31	
	Importance Rating	3.3	

Emergency Procedures / Plan Knowledge of annunciators alarms and indications, and use of the response instructions. (Partial or Complete Loss of Cooling Water)

Proposed Question: Common 45

Unit 1 is at 100% power.

A loss of RBCCW has resulted in the following alarms.

AR-101-B01, RWCU FILTER INLET HI TEMP
AR-101-A01, RWCU FILTER INLET HI TEMP ISO

Which ONE of the following describes the RWCU response and what is the required action IAW OP-161-001 "RWCU Operation"?

A. RWCU INLET OB ISO HV-144-F004 closes.
Immediately throttle open the RWCU Filter Demin Bypass Valve HV-144-F044.

B. RWCU INLET IB ISO HV-144-F001 closes.
Immediately throttle open the RWCU Filter Demin Bypass Valve HV-144-F044.

C. RWCU INLET OB ISO HV-144-F004 closes.
Immediately open RBCCW Supply to RWCU NRHX Valve HV-11315.

D. RWCU INLET IB ISO HV-144-F001 closes.
Immediately open RBCCW Supply to RWCU NRHX Valve HV-11315.

Proposed Answer: A

Explanation (Optional): K/A justification RBCW is the cooling medium for RBCCW

A. Correct.

B. Incorrect - F001 does not auto close under this condition.

C. Incorrect – No procedure guidance to Immediately open HV-11315, it is required to be open after other valve manipulations have occurred.

D. Incorrect - F001 does not auto close under this condition. No procedure guidance to Immediately open HV-11315, it is required to be open after other valve manipulations have occurred.

Technical Reference(s): AR-101-001 A01,Rev.33 (Attach if not previously provided)
AR-101-001 B01, Rev.33

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AA2.02	
	Importance Rating	3.6	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Proposed Question: Common 46

Which ONE of the following describes actions that will occur on a total loss of Instrument Air?

A. The Scram Inlet and Scram Outlet valves fail OPEN.
The Inboard MSIVs and Outboard MSIVs fail CLOSED.
Condensate Short Path Recirc Valve FV-10508 fails OPEN.

B. The Scram Inlet and Scram Outlet valves fail CLOSED.
The Inboard MSIVs fail CLOSED.
Condensate Short Path Recirc Valve FV-10508 fails CLOSED.

C. The Scram Inlet and Scram Outlet valves fail OPEN.
The Outboard MSIVs fail CLOSED.
Condensate Short Path Recirc Valve FV-10508 fails OPEN.

D. The Scram Inlet and Scram Outlet valves fail OPEN.
The Outboard MSIVs fail CLOSED.
Condensate Short Path Recirc Valve FV-10508 fails CLOSED.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The inboard MSIVs are supplied by CIG.
B. Incorrect – Scram Inlet and Outlet Valves fail open. The inboard MSIVs are supplied by CIG. Condensate Short Path Recirc Valve FV-10508 fails open.
C. Correct.
D. Incorrect - Condensate Short Path Recirc Valve FV-10508 fails open.

Attachment A
ON-118-001,Revision 16,Page 15 of 24

ACTIONS THAT OCCUR AS A RESULT OF A LOSS OF I/A PRESSURE
Following events occur in a sequence dependent on location and type of failure(s):
A. CRD - Scram inlet and Scram outlet and backup Scram valves fail OPEN. CRD flow control valve CLOSES to approximately 2% open. Drain and vent valves for Scram Discharge Volume fail CLOSED.
B. Refuel Gates/Cask Pit Gates/Dryer Separator Storage Pit/Refueling Cavity Inflatable Seals - Seals will deflate, causing leakage past gates, which could affect water level in the fuel pool, cask pit and reactor cavity.
C. Emergency Service Water - Supply and return valves for TBCCW and RBCCW heat exchangers fail CLOSED.
D. Condensate - Reject and makeup control valves fail CLOSED. Condensate short path recirculation and condensate pump discharge vent valves fail OPEN.
E. Condensate Filter - Condensate Filter 1 FI 35A-F Inlet and Outlet Valves fail as-is. FV-I 0574A-F fail OPEN and C.F.S. Bypass Valve FV-10572 fails as-is.
F. Main Steam - Outboard MSIVs and deaerating steam supply valves fail CLOSED.

Technical Reference(s): ON-118-001, Rev.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		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295021 AK1.03	
Importance Rating	3.9	

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Adequate core cooling

Proposed Question: Common 47

Unit 1 is in Mode 4 with the following conditions:

- Unit 1 was in Mode 4 with "B" RHR in Shutdown Cooling.
- A reactor coolant leak occurred.
- Shutdown Cooling has isolated.
- RPV level is at + 5 inches.
- RPV temperature is 155 degrees F, rising slowly
- Both Reactor Recirc pumps are shutdown.

Which ONE of the following describes the current status of core circulation and the RPV level requirement IAW ON-149-001 "Loss of Shutdown Cooling"?

A. Core circulation is assured. Maintain level +5 to +35 inches to provide adequate NPSH for subsequent RHR pump start.

B. Core Circulation is assured. Raise level to > +45 inches to ensure adequate water inventory for core boiling.

C. Core circulation is NOT assured. Maintain level +5 to +35 inches to provide adequate level for subsequent Recirc pump start.

D. Core Circulation is NOT assured. Raise level to > +45 inches to provide adequate level for natural circulation.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Core circulation is not assured in the normal level band of + 5 to +35 inches RPV level.

B. Incorrect – Core circulation is not assured at + 5 inches RPV level.

C. Incorrect – RPV level must be raised above 45" to promote natural circulation.

D. Correct – Core circulation is not assured at this level. RPV level must be raised above 45" to promote natural circulation.

Technical Reference(s): ON-149-001, Rev.20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA1.01	
	Importance Rating	3.3	

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : Secondary containment ventilation

Proposed Question: Common 48

Unit 1 is in Refueling with SGTS in a normal alignment. The "A" Refuel Wall Exhaust Radiation Monitor (RISHH-D12-K609A) is determined to be inoperable and untripped. All other systems are operable.

t= 0 Refuel floor accident occurs.

t=1 hr A fire in the "A" Standby Gas Treatment System (SGTS) results in a high-high temperature condition (420 degrees F).

t= 2 hr Related to Refuel Floor conditions, the "B" Refuel Wall Exhaust Radiation Monitor (RISHH-D12-K609B) fails upscale.

How will the SGTS be affected?

A. Both SGTS Trains will start.

B. Neither SGTS Train will start

C. "A" SGTS Train will start
"B" SGTS Train will NOT start.

D. "B" SGTS Train will start
"A" SGTS Train will NOT start.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. The "A" SGT is blocked from starting by the high-high temperature.
B Incorrect, B SGTS will start.
C. Incorrect. The "A" SGT is blocked from starting by the high-high temperature.
D. Correct

Per TM-OP-070, Rev.4

The inlet and outlet of the HECA Beds are monitored for temperature by thermistor- type temperature detectors. These detectors feed a heat detection circuit that will:
— Automatically line up for HECA Bed Cooling if bed inlet temperature increases to 220 °F.
— Start the SGTS Exhaust Fan, and draw cool outside air through the HECA Bed to remove the decay heat when outlet temperature reaches 264 °F.
— Initiate the following actions if inlet or outlet temperature reaches 410 °F.
Tripping the SGTS Exhaust Fan will:
• Close SGTS Crosstie Damper (TD-07560A/B).
• Close SGTS Cooling Outside Air Dampers (HD-07555A/B).
• Close SGTS Inlet Damper (HD-07553A/B).

With the SGTS Exhaust Fan selected to AUTO LEAD, the following conditions result in an automatic fan start.
Start permissives satisfied AND
U-1 or U-2 RPV Low-Low Level (-38 inches), OR
U-1 or U-2 High Drywell Pressure (1.72 psig), OR
U-1 or U-2 Manual Nuclear Steam Supply Shutoff System (N45) Initiation, OR
Refuel Floor Wall Exhaust Duct Radiation High (21 mr/hr), OR
Refuel Floor High Exhaust Duct Radiation High (18 mr/hr), OR
Railroad Access Shaft Exhaust Duct Radiation High (5 mr/hr), OR
SGTS Inlet High Pressure (< 1.5 inches WG), OR
Respective HECA Bed High Temperature (264 °F)
If the fan is selected to AUTO STANDBY, the following conditions initiate an automatic start.
Start permissives satisfied AND
Respective HECA Bed High Temperature (264 °F) with no Zone 3 Isolation Signal present, OR
SGTS Inlet High Pressure (< 1.5 inches WG) with Low Discharge Header Flow after a 30 second time delay, OR
Low Discharge Header Flow (< 2,000 scfm), after a 30 second time delay with a Zone 3 Isolation Signal present.

Technical Reference(s): TM-OP-070, Rev.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 # TMOP070/1991 (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AK1.03	
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Inadvertent criticality

Proposed Question: Common 49

Unit 1 is in refueling. The Refueling operator notices reactor cavity level begin to decrease and places the grappled fuel assembly in an open Reactor location which has a control rod inserted but is NOT the location specified in the FACCTAS.

Reactor cavity level has stabilized and is no longer decreasing.

IAW ON-081-001 "Fuel Handling Accident" the operator has placed the assembly_______.

A. in a preferred location and they must ensure the core remains subcritical by monitoring nuclear instrumentation and have the RE determine shutdown margin.

B. in a preferred location. Shutdown Margin determination is not required, however, the RE must be notified.

C. in a location that is NOT preferred and they must ensure the core remains subcritical by monitoring nuclear instrumentation and have the RE determine shutdown margin.

D. in a location that is NOT preferred. Shutdown margin determination is NOT required, however, the RE must be notified.

Proposed Answer: C

Explanation (Optional): K/A Justification – with the assembly in the wrong location subscriticality and shutdown margin must be determined to ensure an inadvertently criticality has or will not occur.

A. Incorrect – NOT in a preferred location
B. Incorrect – NOT in a preferred location
C. Correct
D. Incorrect – Shutdown margin determination is required

Per ON-181-001, Rev. 8, Section 3.0 & 4.0.

3.1 IF Reactor Cavity/Fuel Pool/Cask Storage Pit level DECREASING and time/radiation levels permit, Insert any grappled fuel assembly into one of the following:

3.1.1 Preferred locations:
a. The FROM location specified on FACCTAS;
b. The TO location specified on the FACCTAS;
c. An open fuel pool location other than a multi-purpose storage container (i.e., gun barrel);
d. An open Dry Storage Canister Fuel location.

3.1.2 If the fuel assembly can not be placed into a preferred location, Insert the assembly into:
a. An open pseudo-cell location in the reactor;
b. An open reactor location which has a control rod inserted.

4.6 IF fuel assembly placed in a core location other than the FROM TO location specified on the FACCTAS:
4.6.1 Ensure core subcritical by monitoring nuclear instrumentation.
4.6.2 Notify Reactor Engineering to determine shutdown margin.

Technical Reference(s): ON-181-001,Rev.8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: __________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

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: Common 50

With Unit 1 at rated conditions, RBCW is lost. The auto swap to RBCCW occurred and drywell pressure is 0.6 psig and rising slowly.

Aside from a reduction in reactor power, minimizing Drywell heat load and pressure rise can be accomplished by which ONE of the following?

A. Placing the operating Drywell coolers in high speed to lower Drywell temperature.

B. Venting the Drywell using SBGT and the two-inch bypass valves to minimize radioactivity release and prevent duct failure in event of a LOCA.

C. Venting the Drywell using SBGT using the 18 inch valves to minimize the duct restrictions and maximize pressure drop rate.

D. Initiating RHR in Drywell spray to lower Drywell pressure and thereby reducing drywell pressure.

Proposed Answer: B

Explanation (Optional):

A	Incorrect. No cooling water is available to the fans. Already running in high speed.
B	Correct.
C	Incorrect. The large valves cannot be used.
D	Incorrect. This is not allowed because you are not yet in the EOPs.

Per AR-104-B03, operator actions include, “Vent Containment in accordance with OP-173-003, Primary Containment Nitrogen Makeup and Venting, IF conditions permit

Technical Reference(s): AR-104-B03, Rev. 24 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # TMOP073
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EA1.07	
	Importance Rating	4.1	

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: ARI/RPT/ATWS: Plant-Specific

Proposed Question: Common 51

Unit 2 was operating at 22% power when the outboard MSIV's shut. The reactor failed to scram. The SRV relief function failed and the SRVs are cycling on their Safety Setpoints.

Which ONE of the following describes how the Reactor Recirculation Pumps are affected by this event?

A. The ATWS-RPT logic trips for both pumps.
The EOC-RPT logic trips for both pumps.
The Recirc MG Drive Motor Breaker does not trip for either pump.

B. The ATWS-RPT logic trips for both pumps.
The EOC-RPT logic does NOT trip for either pump.
The Recirc MG Drive Motor Breaker trips for both pumps.

C. The ATWS-RPT logic does NOT trip for either pump.
The EOC-RPT logic trips for both pumps.
The Recirc MG Drive Motor Breaker trips for both pumps.

D. The ATWS-RPT logic does NOT trip for either pump.
The EOC-RPT logic does NOT trip for either pump.
The Recirc MG Drive Motor Breaker trips for both pumps.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect – The EOC RPT breaker does not trip with Turbine 1st Stage pressure < 123 psig
B. Correct – The pressure transient caused by the MSIV closure and SRV failure will cause reactor pressure to exceed the ATWS-RPT setpoint of 1135 psig. Any RPT trip causes a Drive Motor Breaker trip.
C. Incorrect – The EOC RPT breaker does not trip with Turbine 1st Stage pressure < 123 psig
D. Incorrect - The pressure transient caused by the MSIV closure and SRV failure will cause reactor pressure to exceed the ATWS-RPT setpoint of 1135 psig.

Per TM-OP-064C, Rev.6

Page 40
The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure-High (1135 psig) and two channels of Reactor Vessel Water Level-Low Low, (-38 inches) in each trip

Page 42
The EOC-RPT Breaker Trip occurs on any of the following signals
(provided reactor power is >30 percent as sensed by turbine first-stage pressure of 123 psig):

Technical Reference(s): TM-OP-064C, Rev.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 X
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EK3.02	
	Importance Rating	3.9	

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Proposed Question: Common 52

Which ONE of the following describes the reason for placing RHR in Suppression Pool Cooling as Suppression Pool temperature approaches 90 degrees F in Mode 1?

A. To permit the maximum Suppression Pool temperature limit to be increased to 105 degrees F IAW Technical Specifications during testing which adds heat to the Suppression Pool.

B. To maintain peak primary containment pressure and temperatures within maximum allowable values during a design bases accident.

C. To maintain average containment temperature and relative humidity within established limits for normal plant operations.

D. To minimize ECCS suction strainer and SRV tailpipe quencher thermal stresses during a design bases accident.

Proposed Answer: B

Explanation (Optional):

A	Incorrect - TS allows SP temp to reach 105 °F during testing
B	Correct – Per TS bases
C	Incorrect - Containment humidity not a concern.
D	Incorrect - Strainers and tailpipes not a concern.

Per TS Bases 3.6.2.1, page 3.6.54

A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool.

Technical Reference(s): TS bases 3.6.2.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP049/10497/

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028	
	Importance Rating	3.5	

Common #53

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level measurement.

Which ONE of the following describe the effects of high drywell temperatures on vessel level instrumentation and what instruments are the least susceptible to this effect per EOP Caution 1?

A. Indicated RPV level reads greater than actual.
Wide and Fuel Zone Range level instruments.

B. Indicated RPV level reads greater than actual.
Upset and Extended Range level instruments.

C. Indicated RPV level reads less than actual.
Wide and Fuel Zone Range level instruments.

D. Indicated RPV level reads less than actual.
Upset and Extended Range level instruments.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – Upset and Extended are more susceptible
C. Incorrect – Level reads greater than actual
D. Incorrect – Level reads greater than actual, Upset and Extended range are more susceptible.

Per TM-OP-080-ST

Drywell Temperature
A rise in drywell temperature will cause a rise in temperature of the water in the reference legs of RPV water level instruments. The rise in temperature causes the density of the water in the reference leg to lower. As the temperature of the water in the reference leg rises, the pressure due to the height of water in the reference leg will appear to be smaller. For a given RPV water level, this condition will be observed as a false high level indication. A drop in drywell temperature will have an opposite effect.
Instrument run lengths for reference and variable legs inside the drywell are designed to be nearly the same for Narrow Range, Wide Range, and Fuel Zone Range RPV water level instruments. Therefore, changes in drywell temperatures will have little effect on these indications. Upset, Shutdown, and Extended Range RPV water level instruments are, however, more susceptible to this phenomenon.

Technical Reference(s): TM-OP-080-ST, Rev.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EA2.03	
	Importance Rating	3.7	

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Reactor pressure

Proposed Question: Common 54

Using the attached Heat Capacity Temperature Limit (HCTL) Curve from EO-100-103, "PRIMARY CONTAINMENT CONTROL," SELECT the set of parameters which require Rapid Depressurization.

A. Reactor Pressure: 850 psig
Suppression Pool Temperature: 180 °F
Suppression Pool Level: 18 feet

B. Reactor Pressure: 650 psig
Suppression Pool Temperature: 190 °F
Suppression Pool Level: 15 feet

C. Reactor Pressure: 550 psig
Suppression Pool Temperature: 180 °F
Suppression Pool Level: 15.0 feet

D. Reactor Pressure: 450 psig
Suppression Pool Temperature: 185 °F
Suppression Pool Level: 12.5 feet

Proposed Answer: B

Explanation (Optional):

A	Incorrect. Rapid Depress would be required at 185 °F.
B	Correct.
C	Incorrect. Rapid Depress would be required at 200 °F.
D	Incorrect. Rapid Depress would be required at 195 °F.

Per HCTL Curve on EO-100-103

Technical Reference(s): EO-000-103, Rev.3, HCTL Curve (Attach if not previously provided)

Proposed references to be provided to applicants during examination: HCTL Curve

Learning Objective: ______ (As available)

Question Source: Bank # PP002/2630/
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK 2.14	
	Importance Rating	3.9	

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

Proposed Question: Common 55

The 43CM Local/Remote Control Mode Select Switch has been placed in "Local" at the "B" Diesel Local Control Panel 0C521B.

Which ONE of the following describes the operational status of the "B" Diesel Generator and its associated loads?

The "B" Diesel Generator:

A. must be manually started and the output breaker manually closed by the local operator on either a Loss of Offsite Power (LOOP) or a LOCA signal.

B. must be manually started and the output breaker automatically closes on a LOOP signal. It will NOT automatically start OR load on a LOCA signal

C. must be manually started and the output breaker manually closed by the local operator on a LOOP signal. It will automatically start and load on a LOCA signal.

D. will automatically start and load on either a LOCA or LOOP signal.

Proposed Answer: B

Explanation (Optional): K/A justification – The applicant typically knows that the DGs start on a LOCA (reactor low level/ Hi DW pressure) The question also asks the DGs response to these conditions if in local control.

A. Incorrect – The output breaker automatically closes on a LOOP
B. Correct – The DG must be manually started locally. The output breaker automatically closes on a LOOP
C. Incorrect - The switch in LOCAL removes any auto start functions
D. Incorrect - The switch in LOCAL removes any auto start functions

Per TM-OP-24-ST

Technical Reference(s): TM-OP-24-ST, Rev.8 (Attach if not previously provided)
ON-004-002, Rev.19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank # TMOP024/2256/

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 ______

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EA2.01	
COMMON #56	Importance Rating	4.2	

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

Unit 1 conditions are as follows:

- Reactor power is 25%.
- RPV level is at -75" and steady.
- Nineteen (19) control rods have fully inserted.
- All other control rods are at varying positions.
- When selected, all red 'SCRAM VALVES' lights on the Full Core Display
- are illuminated.
- SDV Vent and Drain valves are closed.
- Both SBLC pumps are injecting

Given the above conditions, which ONE of the following actions would be effective in inserting the control rods and when would the reactor be considered shutdown under all conditions IAW EO-000-113?

A. Venting scram air header.
When SBLC Tank Level drops to 200 gallons.

B. Resetting and scramming again
When SBLC Tank Level drops to 200 gallons.

C. Venting scram air header.
When SBLC Tank level has injected 2800 gallons.

D. Resetting and scramming again.
When SBLC Tank level has injected 2800 gallons.

Proposed Question: Common 56

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Won't work with a hydraulic ATWS
B. Correct – This is below CSBW
C. Incorrect – Won't work with hydraulic ATWS, When the tank level reaches 2800 gallons it is HSBW.
D. Incorrect – When the tank level reaches 2800 gallons it is HSBW

Per EO-000-113 description Page 12

An indicated tank level of 200 gallons means that a volume of boron greater than the Cold Shutdown Boron Weight (CSBW), 4020 gallons, has been injected. CSBW is defined to be the least amount of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions.

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Hot Shutdown Boron Weight (HSBW), 1730 gallons, is defined to be the weight of soluble boron which, if injected into the RPV and uniformly mixed, will maintain the reactor shutdown under hot standby conditions. It assures the reactor will be shutdown irrespective of control rod position when RPV water level is raised to uniformly mix the injected boron.

Technical Reference(s): ON-100-113, Rev.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # PP002/2680/ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295038 EK2.04	
Importance Rating	3.9	

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Stack-gas monitoring system: Plant-Specific

Proposed Question: Common 57

In the event of a high offsite release following fuel degradation, which ONE of the following describes operation of the Post-Accident Vent Stack Sampling System (PAVSSS) and SPING?

PAVSSS_______.

A. provides a standby rad monitoring function to monitor containment vent and purge lines post-accident.

B. provides a redundant and more accurate measurement of stack monitoring, since SPING is for low rad level use.

C. provides redundant monitoring of all ranges when SPING is removed from service for periodic maintenance.

D. provides a backup to SPING, since expected high post-accident background rad levels may result in loss of SPING monitoring capability.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Does not monitor vent and purge lines; Backup for accidents only.
B. Incorrect. PAVSSS can measure similar rad levels as SPING.
C. Incorrect. PAVSSS only has Mid-range noble gas and High-range noble gas channels.
D. Correct. Backup for accidents only.

Page 23
Unlike the SPINGs, where the sample flowrate is adjusted automatically, the PAVSSS flowrate must be adjusted manually. It then delivers the sample stream to the Radiation Detectors. There are only two detectors in service on the PAVSSS; the Mid-Range Noble Gas Channel and the Hi-Range Noble Gas Channel.

Technical Reference(s): TM-OP-079Z-ST, Rev 03 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank # SY017M6/04/1
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 X
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AK3.04	
	Importance Rating	2.8	

Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site

Proposed Question: Common 58

A fire is occurring in the Combo Shop.

What actions are required by ON-013-001 "Response To Fire" and why?

A. Place CREOASS in Emergency Mode to ensure control structure equipment operability.

B. Place CREOASS in Recirc Mode to ensure control room habitability.

C. Place SBGT in Emergency Mode to ensure reactor building equipment operability.

D. Place SBGT in service manually to ensure reactor building habitability.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – the concern per the ON is introduction of smoke into the control room. Placing CREOASS in Emergency will not isolate the control room.
B. Correct – the concern per the ON is introduction of smoke into the control room. Placing CREOASS in recirc isolates the control room. The Combo Shop is located directly across from the new CREOASS intake for the Alternate Source Term modification.
C. Incorrect – SBGT not addressed in the ON.
D. Incorrect – not addressed in the ON

Per ON-013-001, Rev.22, step 5.6.B.1.

Fire outside of the Control Structure could introduce smoke into the Control Room via the outside air plenum. A smoke detector in the plenum will cause an alarm. Placing CREOASS in the recirculation mode will ensure Control Room habitability.

Technical Reference(s): ON-013-001,Rev.22 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295008	
	Importance Rating	3.4	

Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL : Steam flow/feedflow mismatch

Proposed Question: Common 59

A Unit 1 startup is in progress and reactor power is at 60%.

RPV level is 39" and rising.

Which ONE of the following feedwater level control malfunctions is causing the transient?

A. A single steam flow instrument input failing downscale.

B. A single level instrument input failing upscale.

C. A single feed flow instrument input failing downscale.

D. A single feed flow instrument input failing upscale.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – level will go down
B. Incorrect – level will go down
C. Correct
D. Incorrect – level will go down.

Per ON-145-001, Rev.20

2. AUTOMATIC ACTIONS
2.1 If a single level instrument input fails upscale while in "Average", actual RPV water level will stabilize at approximately 22.5".
2.2 If a single level instrument input fails downscale while in "Average", actual RPV water level will stabilize at approximately 47.5".
2.3 If a single steam flow instrument input fails downscale while in Three element", actual RPV water level will stabilize at approximately 25".
2.4 If a single feed flow instrument input fails downscale while in "Three Element", actual RPV water level will stabilize at approximately 48".

Technical Reference(s): ON-145-001, Rev. 20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009 2.1.20	
	Importance Rating	4.3	

Conduct of Operations Ability to execute procedure steps. (Low Reactor Water Level)

Proposed Question: Common 60

The following Unit 1 conditions exist:

- Reactor water level is -10 inches; stable
- Reactor pressure is 600 psig; slowly lowering
- Drywell pressure is 10 psig slowly rising

Which ONE of the following correctly describes how the PCO is required to override RHR Injection IAW OP-149-001?

A. Manually initiate both divisions of RHR; then close the F017A and B Injection Valves after 45 seconds.

B. Close the F017A and B Injection Valves after 45 seconds.

C. Place all RHR Pump Control Switches to stop and verify DGs have cooling.

D. Manually initiate both divisions of RHR; then place all RHR Pump Control Switches to Stop.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. Precaution 2.9.2.a warns that this action will not work >420 psig.
B	Incorrect. Precaution 2.9.2.a warns that this action will not work >420 psig.
C	Incorrect. This only works if the system is currently running; no auto start is present.
D	Correct per Step 2.9.5.

Per OP-149-001

2.9.4 if RHR initiated and RPV pressure > 420 psig, Prevent injection per following:
a. Place pump control switches to STOP and THEN Release.
b. Observe white pump override lights ILLUMINATED, and NO RHR Pumps running.

2.9.5 IF RHR NOT initiated, Prevent injection per following:
a. IF RHR loop potentially voided as determined by ON-137-001:
(1) Close RHR HX A(B) SHELL SIDE INLET HV-151-F047A(B).
(2) Close RHR FIX A(B) SHELL SIDE BYPS HV-151-F048A(B).

b. Arm AND Depress initiation buttons

c. Place pump control switches to STOP, THEN Release.

d. Observe white pump override lights ILLUMINATED, and NO RHR Pumps running.

Technical Reference(s): OP-149-001, Rev.32 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:	Bank #	SY017C01/024/1	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020 AA1.02	
	Importance Rating	3.2	

Ability to operate and/or monitor the following as they apply to INADVERTENT CONT. ISOLATION: Drywell Ventilation/Cooling system

Proposed Question: Common 61

Unit 1 is at 100 percent. The following conditions exist:

- I&C is performing a logic system functional test that trips the "A" Drywell Cooling Isolation Logic Channel.
- After depressing the reset button for the channel, the green light above the manual pushbutton stays lit.
- I&C then inadvertently trips the "C" Drywell Cooling Isolation Logic Channel.

What effect will this have on Drywell Cooling and why?

A. No effect, since only one channel is tripped in each division.

B. No effect, since both tripped channels are in the same division.

C. Both Division 1 and 2 drywell cooling isolation valves close since you have one channel in each division tripped.

D. Only Division 1 drywell cooling isolation valves close since both channels tripped in the same division.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. 'A' and 'C' cause a Division 1 isolation.
B	Incorrect. 'A' and 'C" cause a Division 1 isolation.
C	Incorrect. Only a Division 1 isolation.
D	Correct.

Technical Reference(s): TM-OP-59B, Rev.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # TMOP059B/2142

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK2.07	
	Importance Rating	3.4	

Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor pressure (SCRAM assist): Plant-Specific

Proposed Question: Common 62

A plant startup is in progress.

- All CRD HCU accumulators are operable.
- Reactor pressure has just reached 1,050 psig.
- A loss of all CRD Pumps occurs.
- The mode switch is placed to shutdown.

What is the response of the partially and fully withdrawn CRD Mechanisms?

A. Reactor Pressure alone will insert the control rods fully into the core.

B. Reactor Pressure will start to insert the control rods, and Accumulator pressure will complete the rod insertion.

C. Accumulator Pressure alone will insert the control rods fully into the core.

D. Accumulator Pressure will start to insert the control rods, and the Reactor Pressure will complete the rod insertion.

Proposed Answer: D

Explanation (Optional):

A	A is incorrect. Accumulators will be charged, even with loss of CRD Pump.
B	B is incorrect. Accumulator pressure starts rod scram.
C	C is incorrect. Normally, Rx pressure completes scram.
D	D is correct.

Per TM-OP-055B-ST, page 17

Technical Reference(s): TM-OP-055B-ST, Rev.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # TMOP055/2413

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EK3.02	
	Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Reactor SCRAM

Proposed Question: Common 63

With Unit 1 at 100 percent power. The following conditions exist:

- RWCU Penetration Room area temps are reading 140 °F, rising slowly.
- RWCU Penetration Room ΔTs are reading 50°F, rising slowly.
- RWCU Recirc PP ACC Area Rad Monitor is reading 700 mr/hr, rising.
- RWCU has failed to isolate.

Which ONE of the following actions is required IAW EO-100-104 ?

A. A Primary System is discharging into a Reactor Building area.
Scram the reactor and enter EO-100-102 "RPV Control

B. A Primary System is discharging into a Reactor Building area.
Scram the reactor then Rapidly Depressurize.

C. A Primary System is NOT discharging into a Reactor Building area.
Shut down the reactor IAW GO-100-004 "Plant Shutdown to Minimum Power".

D. A Primary System is NOT discharging into a Reactor Building area.
Continue Power Operation and attempts to isolate RWCU.

Proposed Answer: A

Explanation (Optional):

A. Correct – above max safe with a primary system discharging in the reactor building requires a scram
B. Incorrect – 2 areas are not above max safe, rapid depress is not required
C. Incorrect – A Primary System is discharging Into RB and a scram is required
D. Incorrect - A Primary System is discharging Into RB and a scram is required.

Per EO-000-104, Rev.2, description

Page 17, 18

SC/T-6 **WHEN** RB AREA TEMP EXCEEDS MAX SAFE
IN **2** OR MORE AREAS

SHUT DOWN RX IAW GO-100-004 (GO-200-004)

Should secondary containment area temperature continue to increase and exceed their Max Safe values in more than one area without a primary system discharging into secondary containment, the source of heat addition is most likely a fire. It is prudent to begin a normal reactor shutdown in accordance with GO-100-004 (GO-200-004), Plant Shutdown to Minimum Power. The rapid energy reduction obtained by scramming the reactor will not have an effect on the energy which is causing the increasing temperature trend.

SC/T-7 **BEFORE** ANY RB AREA TEMP REACHES MAX SAFE

1. SCRAM REACTOR
2. GO TO RPV CONTROL

Maximum Safe Operating Levels, or temperatures, are based on three sources:
1) Safety Procedure SP-00-305, 2) isolation setpoints and 3) equipment qualification temperatures. Areas that do not have steam leak detection monitoring are also areas that an Operator routinely accesses on rounds. These areas are not normally above 104°F. SP-00-305 establishes 120°F as the highest temperature for which entry can be made without approval from the Industrial Safety Group. Therefore, Max Safe temperature in areas without temperature monitoring is established at 120°F recognizing the desire to keep normally accessible areas accessible. Entry into high temperature areas during an emergency does not require approval of the Industrial Safety Group.

Other areas are assigned Max Safe temperatures as follows:

Areas monitored by steam leak detection (RWCU equipment, main steam line tunnel, HPCI and RCIC pipe routing, HPCI equipment, and RCIC equipment) are assigned a Max Safe temperature equal to the steam leak detection isolation setpoint. "The setpoints are designed to detect a leakage rate below the leak rate corresponding to critical crack size for the smallest high energy line in the room which is part of the respective system." (FSAR 5.2.5.1.3). Instrumentation and components required for isolation are qualified up to the isolation temperature setpoints. Besides the fact that the equipment will isolate at temperatures above the isolation setpoints, equipment operations is not guaranteed above these temperatures.

Performing a reactor scram in accordance with ON-100-101(ON-200-101), and entry into EO-000-102, RPV Control, promptly reduces to decay heat levels, the energy that the RPV may be discharging to the secondary containment.

Technical Reference(s): EO-000-104 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: E0-100-104 – step SC-4 and below

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295035 EK3.02	
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE : Secondary containment ventilation response

Proposed Question: Common 64

Which ONE of following describes the sequence of how reactor building Zone 1 ventilation is placed in service IAW OP-134-002 “Reactor Building HVAC Zones 1 and 3” and the reason for that sequence?

A. The filtered exhaust fan is started first. Once a negative pressure has been drawn the operator starts the supply and exhaust fans. This sequence minimizes the pressure transient on the building.

B. The filtered exhaust fan is started first. Once a negative pressure has been drawn the operator starts the supply and exhaust fans. This sequence ensures zone drawdown requirements are met.

C. The filtered exhaust fan, the supply fan and the exhaust fan are started simultaneously by placing the control switches for the supply and exhaust fans in start and then starting the filtered exhaust fan. This minimizes the pressure transient on the building.

D. The filtered exhaust fan, the supply fan, and the exhaust fan are started simultaneously by placing the control switches for the supply and exhaust fans in start and then starting the filtered exhaust fan. This sequence ensures zone drawdown requirements are met.

Proposed Answer: C

Explanation (Optional):

A	Incorrect; fans are started simultaneously
B	Incorrect; fans are started simultaneously. Zone drawdown is not a concern for manual startup of the RB ventilation system and is not addressed in the procedure.
C	Correct. Each of these fans is interlocked with the other so the switches are operated in this manner to start all off one switch change.
D	Incorrect; Zone drawdown is not a concern for manual startup of the RB ventilation system and is not addressed in the procedure.

Per OP-134-002,Rev.35, step 2.1.3

NOTE: Zone 1 ventilation system fans are interlocked such that a filtered exhaust fan must be running before supply and exhaust fans can be started. Starting sequence below will allow all three fans to start at the same time to minimize the pressure transient on building.
a. If Reactor Building Zone I HEPA Filter(s) 1 F255A or 1 F255B or 1 F258A or 1 F258B and/or Reactor Building Zone 1 Carbon Filter(s) 1 F257A or 1 F257B is inoperable, Notify Chemistry to perform a dose evaluation per CH-ON-004 Before the Zone 1 Compt. Exh System is placed in service.

Technical Reference(s): OP-134-002, Rev. 35, Step 2.1.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP034/1277
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	500000 EK1.01	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity

Proposed Question: Common 65

Which ONE of the following describes how primary containment is protected against high hydrogen concentrations IAW EO-100-103 "PC Control"?

EO-100-103 "PC Control" requires starting the_______.

A. forced circulation recombiners WHEN H2 concentration reaches 6%.

B. forced circulation recombiners BEFORE H2 concentration reaches 2%.

C. natural convection recombiners WHEN H2 concentration reaches 6%.

D. natural convection recombiners BEFORE H2 concentration reaches 2%.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. Not forced circulation.
B	Incorrect. Not forced circulation.
C	Incorrect. Required before reaching 2% (EOP step PC/G-4)
D	Correct.

Technical Reference(s): EO-000-103, Rev.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 # TMOP073B/10716/ (Note changes or attach parent)

New ______________

Question History: Last NRC Exam ______________

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.19	
Importance Rating	3.0	

Ability to use plant computer to obtain and evaluate parametric information on system or component STATUS

Proposed Question: Common 66

Unit 1 is operating at rated power. The Main Generator is operating with the automatic voltage regulator (AVR) in service. The following parameters currently exist:

- Hydrogen pressure 65 psig
- Generator load 1150 Mwe
- Reactive load +300 MVARs lagging

Refer to the Main Generator Capability Curve (attached) and determine the highest set of generator conditions allowed.

A. 1150 MW and 500 MVARs

B. 1150 MW and 575 MVARs

C. 1200 MW and 275 MVARs

D. 1200 MW and 400 MVARs

Proposed Answer: C

Explanation (Optional):

A	Incorrect - Above the 60 psig curve.
B	Incorrect - Above the 60 psig and 70 psig curve.
C	Correct. - Within the 60 psig curve.
D	Incorrect - above the 60 psig curve.

Technical Reference(s): Main Generator Capability Curve, OP-198-001 Att A, Rev.7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: PICSY display Generator Capability Curve

Learning Objective: ________ (As available)

Question Source: Bank # TMOP098/1163/8

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.30	
	Importance Rating	3.9	

Ability to locate and operate components, including local controls.

Proposed Question: Common 67

The Unit 1 control room has been evacuated due to a potential toxic fume issue.

The plant has been shutdown.
A loss of 1D614 125 VDC has occurred.

You have been directed to operate the ADS SRVs from the field.

Where can the ADS SRVs be operated, if at all, given the above conditions?

A. They can NOT be operated from the Upper Relay Room or Lower Relay Room.

B. They can be operated from Panel 1C628 in the Upper Relay Room.
They can NOT be operated from Panel 1C631 in the Lower Relay Room.

C. They can NOT be operated from Panel 1C628 in the Upper Relay Room.
They can be operated from Panel 1C631 in the Lower Relay Room.

D. They can be operated from either the Upper Relay Room or Lower Relay Room.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – They can be operated from the lower relay room – power is available to that panel.
B. Incorrect - They can be operated from the lower relay room – power is available to that panel. Power was lost to the upper relay room panel.
C. Correct
D. Incorrect – They can only be operated from the lower relay room

Per TM-OP-083E-ST

Page 6

• ADS valves can be operated from Panel 1C628 Upper Relay Room (operates Div I ADS solenoid)
• ADS valves can be operated from Panel 1C631 Lower Relay Room (operates Div II ADS solenoid)

Page 15

125 VDC ELECTRICAL DISTRIBUTION
Panel 1D614 supplies power to the Division I control devices and 1D624 supplies power to the Division II control devices to prevent the loss of one panel from disabling the ADS.

Technical Reference(s): TM-OP-083E,Rev.0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.1	
Importance Rating	3.9	

Knowledge of Conduct of Operations requirements.

Proposed Question: Common 68

A Unit 2 startup is in progress, and reactor power is 21 percent. When withdrawing a control rod to Position 12, the control rod "double-notches" to Position 14. The error is found and corrected when the operator performed the 'reselect and confirm' checks.

Based upon the definition of a Mispositioned Control Rod in ON-155-001, Control Rod Problems, which one of the following is correct?

A. It is a Mispositioned Control Rod, because the control rod motion was not automatically stopped by the Rod Worth Minimizer.

B. It is a Mispositioned Control Rod, because the control rod moved one notch beyond its intended position and was not immediately identified.

C. It is NOT a Mispositioned Control Rod, because the control rod was identified and corrected during the required rod position checks.

D. It is NOT a Mispositioned Control Rod, because the control rod was NOT moved two (2) notches beyond its intended position due to operator error.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Not a determining factor in the definition.
B. Incorrect – One notch in this condition is NOT mispositioned.
C. Correct – The control rod is NOT mispositioned, because the control rod moved only one notch beyond its intended position and was corrected during the required systematic rod position checks.
D. Incorrect – Definition is more than one notch beyond the intended position regardless of why it moved there. (Note: operator versus equipment error, until recently, was a criteria for whether it would be considered a mispositioned rod).

ON-155-001, Rev. 28

1.4 Mispositioned Rod (4)

1.4.1 Control Rod found left in a position other than its intended position and not corrected during "Reselect and Confirm" step of rod movement.
1.4.2 Wrong control rod moved.
1.4.3 Control Rod moved more than one notch beyond its intended position.
1.4.4 Control Rod moved in the wrong direction.

Technical Reference(s): ON-155-001, Rev.28 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # AD044/15315/18
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.2.25	
Importance Rating	2.5	

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: Common 69

Which of the following is the basis for the Technical Specifications, Minimum Suppression Chamber Water Volume in Modes 1, 2, and 3?

A. Ensures a sufficient supply of water is available to provide adequate core cooling and spray the containment with one ECCS Room flooded following a LOCA.

B. Provides a sufficient amount of water to adequately condense the steam from a SRV tailpipe break above the Suppression Pool water level.

C. Ensures a sufficient amount of water would be available to adequately condense the steam from the SRV discharges, downcomers, or HPCI and RCIC turbine exhaust lines and provide emergency make up to the reactor vessel.

D. Provides sufficient supply of water that, along with the Minimum CST Volume, Long-Term Cooling is available following the Design Basis Accident.

Proposed Answer: C

Explanation (Optional): Per TS bases 3.6.2.2

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the S/RV quenchers, downcomers, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

A	Incorrect.
B	Incorrect.
C	Correct - If the Suppression Pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the SRV discharges, downcomers, or HPCI and RCIC turbine exhaust lines. Low Suppression Pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum Suppression Pool water level is specified.
D	Incorrect.

Technical Reference(s): TS Bases 3.6.2.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	TMOP401/13426	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.1.	
	Importance Rating	3.7	

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: Common 70

Which of the following groups of activities is the operator required to perform prior to withdrawing control rods for criticality IAW GO-100-002 "PLANT STARTUP, HEATUP AND POWER OPERATION"?

A. Bypass the Rod Sequence Control System.
Remove Shutdown Cooling from service then place the first Recirc Pump in service.

B. Restore the Rod Sequence Control System.
Remove Shutdown Cooling from service then place the first Recirc Pump in service.

C. Bypass the Rod Sequence Control System.
Start the first Recirc Pump prior to removing Shutdown Cooling from service.

D. Restore the Rod Sequence Control System.
Start the first Recirc Pump prior to removing Shutdown Cooling from service.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – RSCS is bypassed. Recirc must be started prior to securing RHR (a decay heat removal mechanism must be maintained).
B. Incorrect - Recirc must be started prior to securing RHR (a decay heat removal mechanism must be maintained).
C. Correct
D. Incorrect - RSCS is bypassed.

Per GO-100-002

Precautions
3.5 - Do not secure one method of decay heat removal prior to establishing another.

Steps

5.4.7 - Bypass RSCS in accordance with OP-156-001 Section titled “Bypassing/Restoring RSCS”

First control rod is withdrawn at Step 5.22

5.22 - Commence rod withdrawals for criticality as follows:

Technical Reference(s): GO-100-002, Rev.53 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.27	
	Importance Rating	2.6	

Knowledge of the refueling process

Proposed Question: Common 71

Refueling operations are in progress with the reactor vessel head removed and a partial load of fuel is in the vessel.

Which ONE of the following is a CORE ALTERATION?

A. Installing a control rod blade into an empty cell.

B. Driving a Source Range Monitor Detector to full in.

C. Performing a friction test on a control rod in a loaded cell.

D. Inserting the LPRM Instrument Handling Tool below the top guide.

Proposed Answer: C

Explanation (Optional):

A	Incorrect - Control rod movement, provided there are no fuel assemblies in the associated core cell is not considered to be a CORE ALTERATION.
B	Incorrect - Movement of a Source Range Monitor is not considered to be a CORE ALTERATION.
C	Correct - Control Rod movement is a core alteration.
D	Incorrect - The LPRM instrument handling tool is not a reactivity control component.

Per TS 1.1 definitions

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

Technical Reference(s): TS 1.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	TMOP401/2603	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.4	
Importance Rating	2.5	

Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Proposed Question: Common 72

A Susquehanna employee is scheduled to perform a task in an area with the following conditions.

Work Area Radiation Dose Rate: 1060 mr/hr Deep Dose Equivalent (DDE) at 30 cm.
Work Area Contamination: 1900 dpm /100 cm^2 beta-gamma and 35 dpm / l00 cm^2 alpha.

The employee has an accumulated dose of 1700 mr this calendar year.
The work will take the employee 15 minutes to complete while in the area above.

As defined in:
NDAP-QA-0627 "Radioactive Contamination Control,"
NDAP-QA-0626 "Radiologically Controlled Area Access and Radiation Work Permit (RWP) System" and
NDAP-QA-0625 "Personnel Radiation Monitoring Program,

Classify the work area above, and determine whether a dose extension is required due to exceeding annual administrative TEDE limits.

A. This is a High Contamination Area.
This is a Locked High Radiation Area.
A dose extension is NOT required.

B. This is a High Contamination Area.
This is a High Radiation Area.
A dose extension is required.

C. This is a Contamination Area.
This is a High Radiation Area.
A dose extension is required.

D. This is a Contamination Area.
This is a Locked High Radiation Area.
A dose extension is NOT required.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – not a High Contamination area.

B. Incorrect – not a High Contamination area. It is a Locked High Radiation Area.

C. Incorrect – This is defined as a Locked High Radiation Area.

D. Correct – employee does NOT require a dose extension because the administrative limit of 2000 mr TEDE would NOT be exceeded. (Yearly accumulated dose = 1700 mr + 265 mr (15 minutes X 1060mr/hr)= total 1965 mr)

Per NDAP-QA-627,Rev.19

5.3 Contaminated Area - any area, accessible to personnel, in which removable contamination levels are 1000 dpm/100 cm2 beta-gamma or > 20 dpm/100 cm2 alpha.

Per NDAP-QA-626, Rev.5

5.7.1 High Radiation Area - High Radiation Area with dose rates 1 rem/hr at 30 cm from the radiation source or from any surface penetrated by the radiation.

5.7.2 Locked High Radiation Area - High Radiation Area with dose rates > 1 rem/hr at 30 cm from the radiation source OR from any surface penetrated by the radiation, but <500 rads/hr at 1 meter from the radiation source or from any surface penetrated by the radiation.

Per NDAP-QA-625, Rev.9

To ensure compliance with annual occupational radiation dose limits as specified in 10CFR20, radiation dose control levels shall be established for adult radiation workers at Susquehanna, LLC as listed in TABLE 1.

DOSE CATEGORY

Susquehanna, LLC dose shall be controlled, up to:

Total Effective Dose Equivalent (TEDE) 2,000 mrem (20 mSv) without a dose extension

Technical Reference(s): NDAP-QA-0627,Rev.19 (Attach if not previously provided)
NDAP-QA-626, Rev.5
NDAP-QA-0625, Rev.9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.11	
Importance Rating	2.7	

Ability to control radiation releases.

Proposed Question: Common 73

An unisolable steam leak currently exists from the 'A' Main Steam Line in the Turbine Building. The transient resulted in fuel damage. OSCAR Teams have been dispatched to the site boundary.

- All control rods full in
- RPV level is 40 inches and stable with RCIC
- RPV pressure is 800 psig lowering slowly

SPING Noble Gas release value has stabilized at the Alert level. Assume release rate does not change as the RPV is depressurized.

Which ONE of the following is required by EOPs?

A. cooldown at greater than 100 °F/hr using SRVs.

B. cooldown at less than 100 °F/hr using SRVs.

C. cooldown at greater than 100 °F/hr using the Turbine BPVs.

D. cooldown at less than 100 °F/hr using HPCI.

Proposed Answer: B

Explanation (Optional):

A	Incorrect. ADS SRVs are not a rapid depress mechanism.
B	Correct.
C	Incorrect. exceeding cooldown rate is not permitted
D	Incorrect. HPCI operation for pressure control is not permitted with fuel damage

Technical Reference(s): EO-100-102, step RC/P-7 (Attach if not previously provided)

EO-100-105, step RR-5

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # PP002/2680/4
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.4.3	
Importance Rating	3.5	

Ability to identify post-accident instrumentation.

Proposed Question: Common 74

Which ONE of the following groups describes Unit 2 post-accident monitoring (PAM) instrumentation ?

A. Reactor Narrow Range level
Reactor Wide Range level
Reactor Steam Dome (Wide Range) pressure

B. Reactor Wide Range level
Reactor Extended Range level
Reactor Steam Dome (Wide Range) pressure

C. Reactor Narrow Range level
Reactor Extended Range level
Reactor Upset Range level

D. Reactor Fuel Zone Range level
Reactor Shutdown Range level
Suppression Pool Water level

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Not narrow range
B. Correct
C. Incorrect – Not narrow or upset range
D. Incorrect - Not Shutdown range

Per TS 3.3.3.1

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

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.4.5	
Importance Rating	2.9	

Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: Common 75

Unit 1 is operating at 100 percent power:

- RCIC ROOM FLOODED (AR108 H03) annunciates.
- HPCI ROOM FLOODED (AR114 H03) annunciates.
- There is a large leak on the RCIC Pump suction pipe, with ~ 5 feet of water in the Room.
- Water is entering the HPCI Room around the watertight door from RCIC, with ~ 3 feet of water in the Room.

Which one of the following is the required course of action?

A. Enter ON-169-002, Flooding in RB.
Entry to EO-104, SC Control, is NOT required at power.
Commence a normal reactor shutdown and cooldown IAW GO procedure.

B. Enter ON-169-002, Flooding in RB.
Enter EO-104, SC Control, entry into EO-102, RPV Control is NOT required Commence a normal reactor shutdown and cooldown IAW GO procedure.

C. Entry into ON-169-002, Flooding in RB is NOT required.
Enter EO-104, SC Control, and Enter EO-102, RPV Control to Scram, and
Perform a Rapid Depressurization.

D. Entry into ON-169-002, Flooding in RB is NOT required.
Enter EO-104, SC Control and EO-102, RPV Control, to Scram, and
Perform a normal depressurization.

Proposed Answer: B

Explanation (Optional):

A. Entry into EO-104 is required, regardless of plant condition.

B. Correct – Entry into ON-169-002 is required and performed concurrently with EO-104. Since a primary system is not discharging, the plant is shutdown using normal procedures, specifically GO-100-004 procedure. A scram is not required nor desired.

C. Incorrect – Entry into ON-169-002 is required. Entry into EO-102 and/or Rapid Depressurization is not required.

D. Incorrect – Entry into ON-169-002 is required. Entry into EO-102 is not required.

Good KA match since the question requires application of procedure hierarchy (EOP, ON, GO) in accordance with OP-AD-055, OPERATIONS PROCEDURE PROGRAM.

Technical Reference(s): ON-169-002, Rev. 3 (Attach if not previously provided)
EO-000-104, Rev. 2

Proposed references to be provided to applicants during examination: EO-100-104 Table 10 and below SC-4 ONLY.

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # PP002/2680/2 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001 AA2.02	
	Importance Rating		3.2

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

Proposed Question: SRO 76

Unit One is operating at 86% power with both recirculation pumps operating at 80% speed when the "A" recirculation pump trips.

Following the pump trip, the APRMs are showing peak to peak oscillations of three to four percent and steady. No LPRMs are in alarm.

Which ONE of the following describes plant status and actions required?

A. Thermal hydraulic instabilities are occurring. IAW ON-178-002 "Core Flux Oscillations", suppress oscillations by increasing the speed of the "B" recirculation pump.

B. Thermal hydraulic instabilities are occurring. Insert Control Rods IAW RE Instructions in CRC Book to suppress oscillations.

C. Thermal hydraulic instabilities are NOT occurring. IAW ON-178-002 "Core Flux Oscillations", suppress oscillations by increasing the speed of the "B" recirculation pump.

D. Thermal hydraulic instabilities are NOT occurring. Insert Control Rods IAW RE Instructions in CRC Book to suppress oscillations.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Per ON-178-002 – THI is not occurring
B. Incorrect – Per ON-178-002 – THI is not occurring
C. Incorrect – Must decrease pump speed to <80%.
D. Correct

Per ON-178-002,Rev.13
3.3.2 Indications that thermal hydraulic instabilities are occurring by ANY of the following:
a. Peak to peak oscillations <u>trending towards 10% on APRM's</u> (oscillations measured from minimum peak to maximum peak).
b. Peak to peak oscillations trending towards 10w/cm2 on LPRM's.
c. Two (2) or more LPRM upscale lights flashing and clearing on a one to five second period.
d. Two (2) or more LPRM downscale lights flashing and clearing on a one to five second period,
e. OPRM Annunciator Lit

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CAUTION (1)
With one Reactor Recirculation Pump in operation, <u>Rated Pump Speed is limited to ≤ 80% per TRO 3.4.4.</u>
CAUTION (2)
Exceeding the Core Flow Value specified in the CRC Book may cause fuel preconditioning limit violations.
a. Increase the speed of the Operating Recirc PP(s) to suppress oscillations/exit the instability region(s), without exceeding the Core Flow Value in the RE Instructions in the CRC Book.
b. <u>Insert control rods IAW RE Instructions in CRC Book to suppress oscillations</u>/exit the instability region(s).

Technical Reference(s): ON-178-002, Rev.13 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #	295003 2.4.4	
Importance Rating		4.3

Emergency Procedures / Plan Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO 77

A Loss of Offsite Power has occurred and the following conditions exist:

- "A" Diesel is NOT running. There is less than 20 psig in the air receivers.
- "C" Diesel is running loaded to the 1C/2C ESS Buses.
- "B" Diesel tripped on a generator differential. Electrical is investigating.
- "D" Diesel tripped on low lube oil pressure due to a large leak.
- “C” ESW Pump failed to start. All other ESW pumps are available.

Based on this information, which of the following actions is required?

A. Shut down "C" Diesel. Enter EO-100-030 “Station Blackout” and substitute "E" Diesel for "A".

B. Shut down "C" Diesel. Enter EO-100-030 “Station Blackout” and substitute "E" Diesel for "C".

C. Instruct the NPO to reset the "B" Diesel Generator Lockout Relay and substitute "E" Diesel for "B" IAW ON-104-001, “Unit 1 Response to Loss of All Offsite Power”.

D. Allow the "C" Diesel to run and substitute "E" Diesel for "A" IAW ON-104-001, “Unit 1 Response to Loss of All Offsite Power”.

Proposed Answer: A

Explanation (Optional):

A	Correct. Shut down 'C' due to no cooling water and swap 'E' for 'A'. Swap of 'E' for 'A' will provide a complete division.
B	Incorrect. – will not provide a complete division
C	Incorrect. – station blackout procedure applies due to conditions stated (no EDGs running)
D	Incorrect. – station blackout procedure applies due to conditions stated (no EDGs running)

Per EO-100-030,Rev 21

Prior to step 2.10

Contact PCC for an evaluation of system status, and a projected time for power to be restored to the Station.

NOTE:	Preference is for A or B Diesel Generators to be operating when possible.

Step 2.10

When manpower is available, **Substitute** standby Diesel Generator OG501(A)(B)(C)(D)(E), in accordance with OP-024-004, Transfer and Test Mode Operations of Diesel Generator E.

Technical Reference(s): EO-100-030, Rev.21 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source:	Bank #	PP002C/14594/	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295006 AA2.03	
Importance Rating		4.0

Ability to determine and/or interpret the following as they apply to SCRAM : Reactor Water level

Proposed Question: SRO 78

Following a steam leak in Unit 1 Reactor Building, the reactor was manually scrammed. Plant instrumentation reads:

- Drywell Instrument Run Temp is 180°F, slowly rising
- RPV pressure is 150 psig, slowly lowering
- Reactor Building 749' temperature is >250°F and is inaccessible.
- Div 1 and 2 RPV level instruments read similarly.

Division 1 Instruments read:

- WR LI-14201A - 130 inches
- ER LI-14201A - 125 inches
- WR LI-14203A - 130 inches
- ER LI-14203A - 125 inches
- FZ UR-14201A - 155 inches UNCOMP
- FZ UR-14201A - 140 inches COMP

Based upon the above indications, which RPV level and range are reported in accordance with the EOPs and ON-145-004, RPV WATER LEVEL ANOMALY?

A. - 125 inches on Extended Range

B. - 130 inches on Wide Range

C. -140 inches on Fuel Zone

D. -155 inches on Fuel Zone

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Although on-scale, Extended Range is below Minimum Indicated Level and cannot be used.

B. Incorrect – Although on-scale, Wide Range is below Minimum Indicated Level with high Reactor Building Temperatures. Otherwise WR is usable to -145 inches.

C. Correct – Fuel Zone is useable down to -290 inches with these plant conditions. Compensated Fuel Zone off the recorder is reported for RPV level.

D. Incorrect – Uncompensated level is provided by the recorder but is not used for reporting RPV level. Uncompensated level determines whether Fuel Zone can be used and to determine transition from and back to Wide Range. To be reported as actual level, Uncompensated fuel zone would need to be converted manually using the nomograph in ON-145-004 (and this would provide the approximate value of -140 inches at 150 psig, automatically calculated by the compensated level algorithm).

Technical Reference(s): ON-145-004, Rev.12 Attachment C and EOP Caution 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 # PP002/2680/57 (Note changes or attach parent)
- New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295024 2.4.6	
	Importance Rating		4.0

Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies. (High Drywell Pressure)

Proposed Question: SRO 79

During an accident, Unit 1 plant conditions are as follows:

- No Emergency Operating Procedure actions have been taken.
- RPV level - 60 inches; lowering slowly.
- RPV pressure is 90 psig lowering slowly.
- CRD Pump "A" is running.
- Supp Pool level 23 feet.
- Drywell pressure is 19 psig; slowly rising.
- Supp Chamber is 14 psig; slowly rising.
- Drywell temperature is 320 °F; rising.

Which one of the following actions is Shift Supervision required to direct per Emergency Operating Procedures?

A. Enter EO-100-114 "RPV FLOODING" and then Rapidly Depressurize.

B. Enter EO-100-102 "RPV CONTROL", and when level < -205 inches, go to Rapid Depressurization.

C. Enter EO-100-103 "PC CONTROL", initiate Suppression Chamber Spray, limiting flow 1,000 to 2,800 gpm for the first 30 seconds.

D. Enter EO-100-103 "PC CONTROL", initiate DW Spray, limiting flow 1,000 gpm to 2,800 gpm for the first 30 seconds.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. Level can be determined. RPV flooding entry not required
B	Incorrect. Steam cooling not allowed with CRD running.
C	Incorrect. SC Sprays cannot exceed 500 gpm.
D	Correct. DW sprays can be initiated for DW Temp Control w/o doing SC sprays first.

Technical Reference(s): EO-100-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # PP002/2680/

Modified Bank # ____________ (Note changes or attach parent)

New ____________

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025 EA2.04	
	Importance Rating		4.1

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool level

Proposed Question: SRO 80

Following a LOCA and RPV Flooding conditions, the following containment parameters exist:

- UR-15701A(B) Drywell pressure reads 5 psig.
- UR-15701A(B) Suppression Chamber pressure reads 7.5 psig.
- UR-15776A(B) Suppression Pool Level reads 49 feet.

Using ON-159-003, "PRIMARY CONTAINMENT WATER LEVEL ANOMALY", determine containment water level.

Actual containment water level is:

A. 49 feet.

B. 55 feet.

C. 61 feet.

D. 75 feet.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Suppression Pool water level instrument is pegged upscale. In these conditions, ON-159-003 must be used to determine level. DP is 2.5 psid.

B. Correct – A DP of 2.5 psid on Attachment A is 55 feet.

C. Incorrect – This is the result if the operator incorrectly plots 5 psig on Attachment 'A'.

D. Incorrect – This is the result if the operator incorrectly plots 5 psig on Attachment 'B'.

ON-159-003, Rev. 6, Discussion section

5.1 At levels less than 49 feet, Wide Range Suppression Pool Level Indication provides direct readout.

5.2 At levels greater than 49 feet and less than 64 feet, Drywell and Suppression Chamber pressure indication can be used to calculate containment level. At 49 feet the Suppression Chamber Pressure instrument tap becomes submerged, therefore the differential pressure between the Drywell and Suppression Chamber pressure indication is proportional to the height of water above 49 feet. A graph (Attachment A) is provided to determine containment water level based on this differential pressure.

5.3 At levels greater than 64 feet the Drywell Pressure instrument taps are submerged. The height of water can be determined from Drywell pressure indication, however, drywell pressure must be known to ensure the pressure reading is due only to the height of water. Due that all Drywell pressure indication taps are submerged, the only way to ensure that containment pressure is known is if the drywell is vented to atmosphere. With these plant conditions containment water level can be determined. A graph (Attachment B) is provided to determine containment water level based on the height of water at atmospheric pressure. The maximum change in level due to density differences between 100°F and 200°F is approximately 1 foot. Due to the instrumentation and method used this is not discernable. One graph is provided based on 150°F water temperature.

Technical Reference(s): ON-159-003, Rev.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ON-159-003

Learning Objective: __________ (As available)

Question Source:	Bank #	AD045/15307/1	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295037 EA2.04	
Importance Rating		4.1

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

Proposed Question: SRO 81

Unit 1 has experienced a LOCA. Plant auxiliary load shed failed. Plant conditions are as follows:

- Rx Scrammed. Several rods NOT at Position '00'.
- Initial ATWS power is 4.0 percent.
- RPV pressure is 850 psig.
- RPV level is -95 inches and slowly rising with FW.
- SLC is injecting, current tank level is 2,800 gallons.
- Reactor power is on IRM Range 3, and is decreasing.
- Suppression Pool temperature is 195 °F.
- Suppression Pool level is 23.5 feet.
- DW temperature is 260 °F. and slowly rising
- DW pressure is 27.5 psig and increasing.
- Low pressure ECCS overidden.
- RCIC out of service.

Which ONE of the following actions is FIRST required?

A. Stop and Prevent FW injection and Rapidly Depressurize IAW EO-100-112.

B. Commence a cooldown and depressurize at < 100 °F /hr IAW EO-100-102.

C. Restore and Maintain RPV level to +13 inches to +54 inches IAW EO-100-113.

D. Continue FW injection and initiate Rapid Depressurization IAW EO-100-112.

Proposed Answer: A

Explanation (Optional):

A. Correct – HCTL in unsafe region.
B. Incorrect – containment pressure must reach 65#.
C. Incorrect – Must perform rapid depress due to HCTL before raising level.
D. Incorrect – must stop FW injection prior to RD.

Technical Reference(s): EO-100-103 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EO-100-103, HCTL, SP/T leg from SP/T-4 and below, SP/L leg from SP/L -3 and below

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # PP002/2680/68 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	600000 2.1.23	
	Importance Rating		4.0

Conduct of Operations Ability to perform specific system and integrated plant procedures during all modes of plant operation. (Plant Fire On Site)

Proposed Question: SRO 82

A fire has been ongoing in the "A" DG room for the past 10 minutes. No other plant equipment is affected.

Which ONE of the following describes what action is required and how the DGs are credited as a power source for safe shutdown.

A. IAW ON-013-001 "Response to Fire", if CST level instrumentation is not available Transfer Unit 1(One) HPCI suction to the Suppression Pool within 8 hours.
For the Diesel Generator A and C Buildings, offsite power is credited and is used for safe shutdown in place of the diesel generators.

B. IAW ON-013-001 "Response to Fire", if CST level instrumentation is not available Transfer Unit 2 (Two) HPCI suction to the Suppression Pool within 8 hours.
For the Diesel Generator B and D Buildings, offsite power is credited and is used for safe shutdown in place of the diesel generators.

C. IAW ON-013-001 "Response to Fire", if the fire continues for 15 minutes, shutdown the reactor IAW GO-100-004, "Plant Shutdown to Minimum Power."
For the Diesel Generator A and C Buildings, offsite power is credited and is used for safe shutdown in place of the diesel generators.

D. IAW ON-013-001 "Response to Fire", if the fire continues for 15 minutes, shutdown the reactor IAW GO-100-004, "Plant Shutdown to Minimum Power."
For the Diesel Generator B and D Buildings, offsite power is credited and is used for safe shutdown in place of the diesel generators.

Proposed Answer: A

Explanation (Optional):

A. Correct
B. Incorrect – wrong DGs
C. Incorrect not directed in ON-013-001
D. Incorrect – wrong DGs not directed in ON-013-001

Per ON-013-001, page 6

The Purpose of this procedure is to enhance safe shutdown (SSD) capabilities for a worst case fire (Appendix R) in any one fire area. In all areas, except the Diesel Generator A and C Buildings, the emergency diesel generators are credited as the power source for safe shutdown. For the Diesel Generator A and C Buildings, offsite power is credited and is used for safe shutdown in place of the diesel generators. The goals associated with Appendix R Safe Shutdown are as follows:

Attachment I.

IF a fire in A DIG Room: Unit 1 HPCI automatic operation and CST low level signal input to HPCI pump suction automatic transfer logic from CST to Suppression Pool may be lost due to fire damage. Manual transfer of Unit 1 HPCI Pump suction may be required depending on CST level. If CST level instrumentation is not available Transfer Unit 1 HPCI suction to the Suppression Pool within 8 hours.

Technical Reference(s): ON-013-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295013 2.1.12	
	Importance Rating		4.0

Ability to apply tech specs for a system

Proposed Question: SRO 83

Unit 1 is at 80 percent power with SO-152-002, QUARTERLY HPCI FLOW VERIFICATION, in progress.

- At 2200, HPCI was started.
- At 2230, Suppression Pool average water temperature reached 90 °F and rising slowly.
- At 2245, HPCI was tripped as directed in the test procedure and heat addition terminated.
- At 2300, Suppression Pool average water temperature is 92 °F.

What is the latest time that Suppression Pool average water temperature must be less than or equal to 90 °F?

A. 2200 on the next day.

B. 2230 on the next day.

C. 2245 on the next day.

D. 2300 on the next day

Proposed Answer: C

Explanation (Optional):

A. Incorrect – see explanation for correct answer.
B. Incorrect – see explanation for correct answer
C. Correct. The TS action is not required to be entered until the heat addition to the Suppression Pool is terminated, which is when HPCI is tripped per the test procedure (2245). The TS action is not required to be entered when 90 °F is exceeded, because the TS LCO statement is still met, because it can rise to 105 °F during testing that adds heat to the Suppression Pool. TS requires restoring Suppression Pool average water temperature to meet the LCO statement within 24 hours. The TS action is not required to be entered until the heat addition to the Suppression Pool is terminated, which is when HPCI is tripped per the test procedure (2245). The TS action is not required to be entered when 90 °F is exceeded, because the TS LCO statement is still met, because it can rise to 105 °F during testing that adds heat to the Suppression Pool.
D. Incorrect – see explanation for correct answer

Per TS 3.6.2.1

Technical Reference(s): TS 3.6.2.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # SY017M9/2606/26
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295015 AA2.01	
	Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Reactor Power

Proposed Question: SRO 84

Unit 1 was initially at 32% power. An electrical transient resulted in the following conditions:

- Turbine trip occurred.
- Both 'A' and 'B' RPS buses are de-energized.
- Ninety-five (95) control rods failed to insert.
- RPV Level lowered to +20 inches and is +35 inches steady.
- RPV pressure is being controlled on the SRVs.
- All APRMs instruments read downscale.
- All IRMs are reading between 40 and 60 on Range 10.

Which ONE of the following describes the status of reactor power and the EOP actions required?

A. Reactor power cannot be determined.
Enter EO-100-113 at LQ/Q-2 only to insert control rods per EO-100-113 Sheet 2.
SLC injection is required.

B. Reactor power is less than 5 percent.
Enter EO-100-113 at LQ/Q-2 only to insert control rods per EO-100-113 Sheet 2.
SLC injection is not required.

C. Reactor power is greater than 5 percent.
Enter EO-100-113 at LQ-1 to intentionally lower RPV level to below -60 inches.
SLC injection is required.

D. Reactor power cannot be determined.
Enter EO-100-102, RPV Control to maintain RPV level between +13 and +54 inches. SLC injection is not required.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Reactor Power is >5%. Eo-113 actions to lower level.
B. Incorrect – Reactor Power is >5%. Eo-113 actions to lower level and inject SLC required
C. Correct.
D. Incorrect – Reactor Power is >5%. Eo-113 actions to lower level and inject SLC required

Technical Reference(s): EO-100-102 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295017 2.4.29	
	Importance Rating		4.0

2.4.29 – Emergency Procedures/Plan: Knowledge of the emergency plan (High Offsite Release Rate)

Proposed Question: SRO 85

Following a transient, a confirmed sample analyses for gaseous release indicates the following for total site release:

The following sets of data are provided at the times specified.

Time 12:00

Noble gases @ 1.1 mrem/year whole body

Noble gases @ 4.7 mrem/year skin

1-131, 1-133, H-3, and particulates with half-lives >8 days @ 2,500 mrem/year to any organ (inhalation pathways only).

Time 12:10

Noble gases @ 8.9 E+4 mrem/year whole body

Noble gases @ 6.1 E+5 mrem/year skin

1-131, 1-133, H-3, and particulates with half-lives >8 days @ 2,900 mrem/year to any organ (inhalation pathways only).

Time 12:30

Noble gases @ 9.4 E+4 mrem/year whole body

Noble gases @ 6.7E+5 mrem/year skin

I-131, 1-133, H-3, and particulates with half-lives >8 days @ 2.8E+5 mrem/year to any organ (inhalation pathways only).

Which ONE of the following describes the EAL classification?

A. An Unusual Event was reached at 12:00

B. An Alert was reached at 12:00

C. An Alert was reached at 12:25

D. A Site Area Emergency was reached at 12:30

Proposed Answer: C

Explanation (Optional):

A. Incorrect – the alert level readings must be for at least 15 minutes
B. Incorrect – An Alert was reached at 12:30
C. Correct
D. Incorrect – An Alert was reached at 12:30. No SAE level was reached.

PER TM-OP-ST-079Z, Rev.1, page 35 & 36. EALs 15.2.a.

EMERGENCY CLASSIFICATION BASED ON EXCEEDING TECHNICAL REQUIREMENTS INSTANTANEOUS RELEASE RATES

Unusual Event

Radiological effluents exceed Technical Requirements Limits as indicated by:

- Valid Building Vent Stack Monitoring System (SPING) indications on Panel 0C630 or 0C677.
- Noble gases, > 1.70E+6 iCi/min.,or
- 1-131, >2.08E+2 iCi/min.,or
- Particulate >1 .54E+3 iCi/min or

OR

- Confirmed sample analyses for gaseous releases indicating total site release rates exceed:
- Noble gases >1,000 mrem/year whole body, or
- Noble gases >6,000 mrem/year skin, or
- 1-131, 1-133, H-3, and particulates with half-lives >8 days >3,000 mrem/year to any organ (inhalation pathways only).

Alert

EAL# 15.2.a Radiological gaseous effluents exceed 200 times the Technical Requirement Limits for 15 minutes or longer as indicated by:

- Valid Building Vent Stack Monitoring System (SPING) indications on Panel 00630 or 00677.
- Noble gases, > 1.70E+8 iCi/min or
- 1-131, >2.08E+4 iCi/min.,or
- Particulate >1 .54E+5 iCi/min or

OR

- Confirmed sample analyses for gaseous releases indicating total site release rates exceed:
- Noble gases >1 .OE+5 mrem/year whole body, or
- Noble gases >6.OE+5 mrem/year skin, or
- 1-131, 1-133, H-3, and particulates with half-lives >8 days >3.OE+5 mrem/year to any organ (inhalation pathways only).

Technical Reference(s): EALs (Attach if not previously provided)
EP-TP-001, Table R,
Attachment R (page 17 of 186).

Proposed references to be provided to applicants during examination: Table R, Attachment R of EP-TP-001

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	203000 2.4.30	
	Importance Rating		3.6

Emergency Procedures / Plan Knowledge of which events related to system operations/status should be reported to outside agencies. (RHR System)

Proposed Question: SRO 86

Unit 1 is at 20% power. During post maintenance testing, the Unit 1 RHR Inboard injection valve HV-151-F015A failed to open. Maintenance reports that the valve is mechanically bound and cannot be opened.

Stroke testing on the RHR Inboard injection valve HV-151-F015B resulted in the same problem and the valve failed to open.

What are the Technical Specification requirements and what notification must be made?

(For the required notification consider only the Technical Specification action)

A. Immediately enter TS 3.0.3.
1 hour ENS notification is required.

B. Immediately enter TS 3.0.3.
4 hour ENS notification is required.

C. De-energize the F015A and F015B within 1 hour and restore One loop of RHR within 7 days.
8 hour ENS notification is required.

D. De-energize the F015A and F015B within 1 hour and restore One loop of RHR within 7 days.
24 hour ENS notification is required.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – A 4 hour notification is required due to the TS required shutdown. 1 hour is not required per the TS.
B. Correct
C. Incorrect – Valves must be deactivated within 4 hours. TS requires entry to 3.03 with both loops inoperable for the ECCS function.
D. Incorrect – Valves must be deactivated within 4 hours. TS requires entry to 3.03 with both loops inoperable for the ECCS function. 24 hours is not correct for this situation.

Per NDAP-QA-0720, Rev.2, Att.F

FOUR HOUR ENS NOTIFICATIONS
Initiating Event
1. SHUTDOWN REQUIRED BY THE TECHNICAL SPECIFICATIONS
Reference
Additional Reporting Req.
The initiation of any nuclear plant shutdown required by the plant's Technical Specifications. (Call to be made upon physical commencement of power reduction.)
1 0CFR5072(a)(1)(ii)
1 OCFR5O.72(a)(5)(ii)
1 OCFR5O.72(b)(2)(i)
1 OCFR5O.72(c)
1 OCFR5O.36(c)(1)

TS 3.5.1.I requires entry into LCO 3.03 immediately

Technical Reference(s): TS 3.5.1.I (Attach if not previously provided)
NDAP-QA-0720
TS 3.6.1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History:	Last NRC Exam	______
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	______
10 CFR Part 55 Content:	55.41	______
	55.43	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	223002 2.1.9	
	Importance Rating		4.0

Conduct of Operations - Ability to direct personnel activities inside the control room. (PCIS/N4S)

Proposed Question: SRO 87

A plant startup is in progress on Unit 1. The following conditions exist:

- Reactor power is three percent.
- Reactor pressure is 920 psig.
- Reactor level is +35 inches.
- The Mechanical Vacuum Pump is in service; Main Condenser vacuum is 7 inches HgA.
- The Main Steam Line Line Rad Monitors readings are slightly elevated..
- Two Condensate Pumps and one RFP are in service.
- Turbine Building Steam Tunnel temperature is 201 °F and rising.
- The MSIVs are open.

Which ONE of the following must you direct the ROs to perform/verify and what procedure entries are required?

A. Close the MSIVs. No Scram is required. Refer to ON-184-001, Main Steam Line Isolation and Quick Recovery

B. Scram the reactor and Close the MSIVs. Refer to ON-184-001, Main Steam Line Isolation and Quick Recovery and ON-100-101 " Scram, Scram Imminent."

C. Trip the Mechanical Vacuum pump. No Scram is required. Enter ON-143-001 "Main Condenser Vacuum and Off-Gas System Off Normal Operation."

D. Scram the reactor and trip the Mechanical Vacuum pump. Enter ON-143-001 "Main Condenser Vacuum and Off-Gas System Off Normal Operation" and ON-100-101 "Scram, Scram Imminent."

Proposed Answer: B

Explanation (Optional):

A. Incorrect – increasing pressure will cause need to manually scram before automatic scram at 1087.
B. Correct – no automatic scram occurs at 3% power on MSIV closure, manual scram required to prevent high pressure automatic scram
C. Incorrect – vacuum pumps unaffected
D. Incorrect – vacuum pumps unaffected. No scram occurs.

Technical Reference(s): ON-184-001,Rev.10 (Attach if not previously provided)
AR-111-001,Rev.32

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____ (As available)

Question Source: Bank # TMOP059B/2142
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	239002 A2.02	
	Importance Rating		3.2

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leaky SRV

Proposed Question: SRO 88

Unit 1 is 100% power, all systems operable, when the following are observed.

- Reactor Power shows slight increase
- Reactor Pressure is decreasing
- Feedwater Temperature is decreasing
- Generator MWE is decreasing

Which ONE of the following describes what has occurred and the procedure entry required.

A. The backup EHC pressure regulator has taken control. Enter ON-193-001 "Turbine EHC System Malfunction".

B. An SRV has opened. Enter ON-183-001 "Stuck Open Safety Relief Valve".

C. A Control Rod with a high rod worth has drifted in. Enter ON-155-001 "Control Rod Problems".

D. Extraction Steam to a feedwater heater string has isolated. Enter ON-147-002 "Loss of Feedwater Heater String".

Proposed Answer: B

Explanation (Optional):

A. Incorrect – this would cause an increase in reactor pressure
B. Correct
C. Incorrect – reactor power would decrease but would stabilize,
D. Incorrect – reactor power would increase. A loss of extraction steam requires entry to ON-147-001

Per ON-183-001,Rev.23, page 2

SYMPTOMS AND OBSERVATIONS
1.1 Alarms at Reactor Core Cooling Systems Benchboard 1C601:
1.1.1 MAIN STEAM SRV LEAKING (AR110-E01)
1.1.2 SUPP POOL DIV 1 AVERAGE TEMP HI (AR111-F04)
1.1.3 SUPP POOL DIV 2 AVERAGE TEMP HI (AR112-E04)
1.1.4 MAIN STEAM DIV 1 SRV OPEN (AR110-E02)
1.1.5 MAIN STEAM DIV 2 SRV OPEN (AR110-E03)
1.2 <u>Indicated feedwater flow greater than indicated steam flow.</u>
1.3 <u>Loss of generator MWE</u>.
1.4 <u>Feedwater temperature decrease</u> due to SRV steam bypassing feedwater heating.
.5 <u>RPV pressure decreasing.</u>
1.6 RPV level swell.
.7 Suppression pool temperature increasing.
1.8 Suppression pool level increasing.
.9 Suppression chamber pressure increasing.

Technical Reference(s): ON-183-001 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	264000 A2.06	
Importance Rating		3.4

Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Opening normal and/or alternate power to emergency bus

Proposed Question: SRO 89

The following conditions exist on Unit 1:

- MODE 1, 100 percent RTP.
- Transformer T-101 (0X201) is out of service.
- 4.16 kV Buses are energized.

Subsequently, Transformer T-201 (0X203) Lockout Relay actuates. All other equipment operates as designed.

What Unit 1 procedural entries are required?

A. ON 104-201 "Loss of 4KV Bus 1A and ON 104-204 "Loss of 4KV Bus 1D for a Momentary Loss of Bus

B. ON 104-201 "Loss of 4KV Bus 1A and ON 104-204 "Loss of 4KV Bus 1D for a Sustained Loss of Bus.

C. ON 104-202 "Loss of 4KV Bus 1B and ON 104-203 "Loss of 4KV Bus 1C for a Momentary Loss of Bus

D. ON 104-202 "Loss of 4KV Bus 1B and ON 104-203 "Loss of 4KV Bus 1C for a Sustained Loss of Bus.

Proposed Answer: A

Explanation (Optional):

A. Correct – Momentary loss because the EDGs pick up the Bus
B. Incorrect – Not a Sustained loss.
C. Incorrect – those buses are being supplied by the opposite ESS transformers
D. Incorrect - those buses are being supplied by the opposite ESS transformers

Technical Reference(s): ON-104-201,Rev.9 (Attach if not previously provided)
ON-104-204,Rev.8

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	400000 2.1.33	
	Importance Rating		4.0

Conduct of Operations Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (Component Cooling Water System)

Proposed Question: SRO 90

Unit 2 is operating at 100 percent power. Unit 1 scrammed eight hours ago and a cooldown is now in progress with reactor pressure at 105 psig.

The outside NPO reports that the Spray Pond level has lowered since the scram and is now 678.0 feet, lowering.

Which ONE of the following describes Unit 1 (One) Tech Spec Operability?

A. ONLY one ESW subsystem is declared Inoperable.
RHRSW is Operable
Ultimate Heat Sink is Operable.

B. ESW is Operable.
ONLY one RHRSW subsystem is declared Inoperable.
Ultimate Heat Sink is Operable.

C. BOTH ESW subsystems are declared Inoperable.
RHRSW is Operable.
Ultimate Heat Sink is declared Inoperable.

D. ESW is Operable.
Both RHRSW subsystems are declared Inoperable.
Ultimate Heat Sink is declared inoperable.

Proposed Answer: D

Explanation (Optional):

A	Incorrect - Per TS Bases, UHS is bounded by RHRSW spec, ESW not directly impacted.
B	Incorrect - Both RHRSW subsystems are Inoperable.
C	Incorrect - Per TS Bases, UHS is bounded by RHRSW spec, ESW not directly impacted.
D	Correct. LCO 3.7.1 must be entered since Spray Pond level is low (< 678 feet, 1 inch), and per bases, required for Operable RHRSW Subsystem.

Technical Reference(s): TS Bases 3.7.1 and 3.7.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP016/12545/1

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	215002 2.2.25	
	Importance Rating		3.7

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (RBM)

Proposed Question: SRO 91

The basis for the Unit 2 Rod Block Monitor (RBM) Technical Specification LCO ensures that the RBM will function to ___________.

A. prevent fuel clad damage resulting from a startup Rod Withdrawal Error in the Intermediate Range.

B. prevent fuel clad damage resulting from an out-of-sequence control rod withdrawal during a startup.

C. provide PCI and backup LHGR protection to prevent high local powers during rod withdrawals at power.

D. provide primary MCPR protection to prevent high local power during a Rod Withdrawal Error at power.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. RBM not active during startup conditions, until LPSP <30%
B	Incorrect. RBM does not protect against an out-of-sequence control rod withdrawal.
C	Incorrect. On Unit 1, RBM presently not credited in the RWE analysis, but provides backup MCPR protection. On Unit 2, with EPU, the RBM is credited and required to protect MCPR for the RWE.
D	Correct.

Unit 1 TS Bases 3.3.2.1
The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The RBM was originally designed to prevent fuel damage during a Rod Withdrawal Error (RWE) event while operating in the power range in a normal mode of operation. FSAR 15.4.2 (Ref. 10) (Rod Withdrawal Error - At Power) originally took credit for the RBM automatically actuating to stop control rod motion and preventing fuel damage during an RWE event at power. However, current reload analyses do not take credit for the RBM system

Unit 2 TS Bases 3.3.2.1
The RBM is designed to prevent violation of the MCPR SL and the cladding 1% strain fuel design limit that may result from a Single Control Rod Withdrawal (RWE) event.

Technical Reference(s): TS Bases 3.3.2.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________________ (As available)

Question Source:	Bank #	TMOP078K/1544/2	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	259001 A2.03	
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of condensate pump(s)

Proposed Question: SRO 92

Unit 1 is operating at 100% power when ONE of the operating Condensate Pumps trip.

The following conditions exist:

- Reactor Recirc Pump Runback circuit fails to actuate.
- Feedwater Pump Suction Pressure has lowered to 250 psig for the past 11 seconds.

Which ONE of the following describes the response of the feedwater system and procedure entry required as a result?

A. Only “A” Feedwater Pump would trip due to this low suction pressure condition. Scram the reactor and enter ON-100-101 “Scram - Scram Imminent.”

B. “A” and “B” Feedwater Pumps would trip due to this low suction pressure condition. Enter GO-100-012 “Power Maneuvers” to reduce power to within feedwater pump capability.

C. “A” and “B” Feedwater Pumps would trip due to this low suction pressure condition. Scram the reactor and enter ON-100-101 “Scram - Scram Imminent.”

D. Only “A” Feedwater Pump would trip due to the low suction pressure condition. Enter GO-100-012 “Power Maneuvers” to reduce power to within feedwater pump capability.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – "A" and "B" FW pumps will trip
B. Incorrect – Entry to Scram ON is required
C. Correct
D. Incorrect – "A" and "B" FW pumps will trip. Entry to Scram ON is required

RFP Suction Pressure
The suction pressure for each RFP is continuously monitored and recorded on Control Room Recorder PR-10609 located on Panel 10652. In addition, PT-10610A/B/C provides input into the respective RFPT trip logic. The setpoint for the low suction pressure trip is 264 psig. To prevent a transient from tripping all RFPTs, the trips are time delayed as follows:

• RFP 'A' - 264 psig with a five-second time delay
• RFP 'B' - 264 psig with a 10-second time delay
• RFP 'C' - 264 psig with a 15-second time delay

Technical Reference(s): AR-101-001,Rev.33 (A10,A12) (Attach if not previously provided)
GO-100-012,Rev.24

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	271000 A2.04	
	Importance Rating		4.1

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 high radiation

Proposed Question: SRO 93

With a Unit 1 Turbine Building SPING alarm alarming from a valid high radiation signal.

How can it be definitively determined if the source of radiation was from Radwaste IAW ON-070-001 "Abnormal Gaseous Radiation Release." and what actions would be required for a valid alarm?

A. Close one pair of OFFGAS DELAY LINE DRAIN VLVS while monitoring Turbine Building exhaust radiation levels IAW OP-172-001 "SJAE and Offgas System.

B. Verify Unit 1 Turbine Building exhaust radiation levels decrease when Radwaste ventilation is secured IAW OP-065-002, Radwaste Building HVAC.

C. Direct HP to check the Decon Shop. Shutdown the Decon Shop HVAC.

D. Secure the Charcoal Adsorber HVAC units IAW OP-172-003 "Operation of Off Gas Charcoal System".

Proposed Answer: B

Explanation (Optional):

A	Incorrect. This exhaust to the TB Vent SPING Monitoring System via TB HVAC. This might help, but it is not the definitive answer and is directed in the ON for the TB.
B	Correct.
C	Incorrect. This is part of Turbine Building HVAC.
D	Incorrect. This will elevate the Turbine Building release rate and provide no definitive indication the airborne is coming from radwaste.

Per ON- 070-001, Rev.14

3.5.4 Determine if source from Radwaste Building as follows:
a. Shut Down Radwaste Building Ventilation System in accordance with OP-O65-002, Radwaste Building HVAC.

Technical Reference(s): ON- 070-001,Rev.14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # AD045/1358/52
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.14	
	Importance Rating		3.3

Knowledge of system status criteria which require the notification of plant personnel

Proposed Question: SRO 94

During a plant startup, the "A" Recirc MG Set has tripped. You have completed the OI-AD-066, PLANT SYSTEM/COMPONENT TROUBLE REPORTING, Attachment R.

Which ONE of the following is shift supervision required to ensure IAW OI-AD-066?

That the proper attachments of the OI are _________.

A. emailed to the System Engineer.

B. submitted to FIN for investigation.

C. filed in the OI-AD-066 book in the STA Office.

D. attached to the AR and forwarded to Relay and Test.

Proposed Answer: D

Explanation (Optional):

A	Incorrect - NSE receipt not required.
B	Incorrect. FIN will investigate after they get the AR.
C	Incorrect - This is not required by the OI.
D	Correct - This is the standard process OI-AD-066.

Per OI-AD-066

2.1 Shift Supervision

Ensures proper attachments of this instruction accompanies all applicable ARs issued. (Attachment can be scanned and electronically attached to AR, or forwarded to OES.)

4.4 Reactor Recirculation MG Set Control Panel Electrical Target Reporting.

4.4.1 Maintain blank copies of Attachment R, Form OI-AD-066-18 at both Unit 1 and 2 Recirc MG control panels, 1/2C063A and B.

4.4.2 Complete Attachment R for each Recirc MG Set trip at time of trip, to be used to identify targets.

4.4.3 Forward completed copy of Attachment R to Relay and Test and attach to AR.

Technical Reference(s): OI-AD-066 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source:	Bank #	AD044/14879	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.11	
	Importance Rating		3.8

Knowledge of less than one hour technical specification action statements for systems.

Proposed Question: SRO 95

Unit 1 is at 100 percent reactor power.
I&C notified Operations of a calibration error on the "D" MSL Flow-High isolation instrumentation. The trip setpoint data is as follows:

D MSL Flow-High Instrument Number	Trip Setpoint
FIS-B21-1N009A	117 psid
FIS-B21-1N009B	118 psid
FIS-B21-1N009C	125 psid
FIS-B21-1N009D	123 psid

The allowable value per TS table 3.3.6.1-1 for the trip setpoint is ≤ 121 psid.

What Technical Specification required action and completion time, if any, is applicable at the time of discovery?

A. None, LCO is met.

B. Place required channel(s) in trip within 24 hours.

C. Restore Isolation capability within one hour.

D. Be in MODE 2 in seven hours.

Proposed Answer: B

Explanation:

A. Incorrect - LCO is not met, Table 3.3.6.1-1 requires each trip system to have 2 channels/main steam line operable.

B. Correct - A and B channels are operable. One channel is inoperable in each trip system. Isolation capability is maintained but the inop channels will be required to be placed in the tripped condition within 24 hours.

C. Incorrect - Would be correct if candidate believes that isolation capability is not maintained.

D. Incorrect - Time from LCO 3.0.3 which is not applicable.

The issue of calibration error adds complexity since have to make decision if equipment is broke or not. Have to determine from data if sufficient number of instruments and then determine if function is available.

Technical Reference(s): Unit 1 TS 3.3.6.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # TMOP083/12699/1

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______

55.43 X

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		
K/A #	2.2.5	
Importance Rating		2.7

Knowledge of the process for making changes in the facility as described in the safety analysis report

Proposed Question: SRO 96

As outlined in NDAP-QA-0726, "10 CFR 50.59 and 10 CFR 72.48 Implementation," which ONE of the following proposed changes would require a 10 CFR 50.59 Screen?

A. Maintenance and refurbishment of the RHR Injection Flow Control Valve HV-151-F017A limitorque actuator.

B. Add procedure section to bypass trip setpoints on the Refuel Floor Wall Exhaust Duct Rad monitoring instrument.

C. Moving the Security perimeter fence to include the entire 500kV yard as part of the onsite facility protected area.

D. Relocating the TSC emergency response facility from the Control Structure to the West Building.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Not applicable, normal maintenance evolution restores to original design configuration.

B. Correct - Refuel Floor Wall Exhaust Duct Rad Monitor setpoints are required for accident mitigation and SSC operation.

C. Incorrect – Not applicable since outside scope of 50.59; Security systems and designs are regulated by 10 CFR 73.

D. Incorrect – Not applicable since outside scope of 50.59 screen; Emergency Plan facilities are regulated by 10 CFR 50.47.

NDAP-QA-0726, Rev, 10
Attachment A, Applicability Determination

Technical Reference(s): NDAP-QA-0726, Rev.10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	NDAPQA0726/16	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		
K/A #	2.3.3	
Importance Rating		2.9

Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems).

Proposed Question: SRO 97

A liquid radioactive waste (LRW) Sample Tank release is required to reduce LRW Excess inventory. The Radwaste Effluent Rad Monitor is out of service.

Who must approve the discharge, and what actions are required in order to commence the discharge IAW NDAP-QA-0310?

A. The Field Unit Supervisor must approve the discharge. A minimum of two independent samples must be obtained and analyzed from the LRW Sample Tank.

B. The Field Unit Supervisor must approve the discharge. Discharge must be estimated once per four hours during the discharge.

C. The Shift Manager must approve the discharge. Discharge must be estimated once per four hours during the discharge.

D. The Shift Manager must approve the discharge. A minimum of two independent samples must be obtained and analyzed from the LRW Sample Tank.

Proposed Answer: D

Explanation (Optional):

A	Incorrect. SM approval required
B	Incorrect. SM approval required.
C	Incorrect. Estimation not required , only 2 independent samples
D	Correct.

Per NDAP-QA-0310, Rev. 5

RESPONSIBILITIES

4.1 Shift Manager

4.1.1 Final authorization prior to commencement of liquid effluent releases.

Technical Reference(s): NDAP-QA-0310 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank # AD044/4696/
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.3.9	
	Importance Rating		3.4

Knowledge of the process for performing a containment purge.

Proposed Question: SRO 98

Unit 1 conditions are as follows:

- A large break LOCA has occurred.
- RPV level is -110" AND steady.
- Primary Containment combustible gas concentrations are rising slowly.

The TSC has directed that the drywell be purged in accordance with ES-173-005, PURGE CONTAINMENT WITH NITROGEN WITHIN TECH SPEC RELEASE LIMITS, to lower containment combustible gas concentration.

Which of the following describes the location and method identified in ES-173-005 for bypassing PCIS interlocks for subsequent opening of Drywell Vent and Purge Valves:

	24" DRWL VENT IB ISO HV-15713	24" DRWL VENT OB ISO HV-15714	2" DRWL VENT BYPS OB ISO HV-15711
A.	Main control room switch	Main control room switch	Lower relay room jumper
B.	Lower relay room jumper	Lower relay room jumper	Main control room switch
C.	Lower relay room jumper	Main control room switch	Lower relay room jumper
D.	Main control room switch	Lower relay room jumper	Main control room switch

Proposed Answer: D

Explanation (Optional):

A. Incorrect – OUTBD valve HV-15714 requires jumper installation. Plant design prevents bypass and operation of large purge and vent valve flowpath with an isolation signal present. Requires EOP, ES and jumpers.

B. Incorrect – INBD valve HV-15713 provided with control room switch for 2" flowpath.

C. Incorrect – OUTBD valve HV-15711 provided with control room switch for 2" flowpath.

D. Correct – ES-173-005 directs that the interlocks for the 24" INBD, HV-15713 and the 2" OUTBD, HV-15711 be bypassed using the bypass switch on 1C651. The interlocks for the 24" OUTBD, HV-15714 are NOT equipped with a bypass switch due to flowpath and the ES procedure requires the installation of a jumper in 1C661B3.

TM-OP-073, Rev.01 pages 27-28
These valves open when their control switch is placed in OPEN position provided that none of the close signals are present. They CLOSE automatically on:
PCIS signal (62X4-K59B) from the RHR logic:

- Low-Low Reactor Pressure Vessel (RPV) Level 2 < -38"H2O
- High Drywell Pressure > 1.72 psig
- Manual Isolation pushbutton (switch HS14142A (B) in ARMED position)
  AND
- Test Mode (switch HS14143A (B) in TEST LOCA position)
- Drywell Purge Line to SBGT High Pressure (PSH15704 for Suppression Chamber valve) or (PSH15714 for Drywell valve) at 0.2 psig to prevent SBGT system over pressure
- High SBGT Exhaust Radiation Level > 23mR/hr (RISHHD12K617B)

These valves will remain closed until the condition clears and the isolation signal is RESET. There are no override switches for these valves. If it becomes necessary to open the valves, as directed by Emergency Support procedures, the isolation signals can be overridden manually using jumpers.

Technical Reference(s): TM-OP-073, Rev. 01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source:	Bank #	TMOP073/1882/7	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.9	
	Importance Rating		3.9

Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies

Proposed Question: SRO 99

Unit 1 is in Mode 4 and RHR is in Shutdown Cooling. Primary and Secondary Containment is NOT established.

IAW ON-149-001, which ONE of the following is considered a permitted alternate method of decay heat removal if a Loss of Shutdown Cooling were to occur and what is a concern associated with the method used?

A. Core Spray injection from suppression pool and return path through 2 SRV's (PSV-141-F013C,E,F,L,M or R preferred) IAW OP-151-001 Core Spray System.
A concern in this configuration is that the 100°F/hour cooldown rate may be exceeded.

B. Core Spray injection from suppression pool and return path through 2 SRV's (PSV-141-F013C,E,F,L,M or R preferred) IAW OP-151-001 Core Spray System.
A concern is inadequate heat removal capabilities.

C. Allow reactor pressure to rise to >20 psig but less than 98 psig and route the condensing steam to the main condenser IAW OP-144-001 Condensate and Feedwater System.
A concern is release of radioactive materials to the environment.

D. Allow reactor pressure to rise to >20 psig but less than 98 psig and route the condensing steam to the main condenser IAW OP-144-001 Condensate and Feedwater System.
A concern is inadequate heat removal capabilities.

Proposed Answer: A

A. Correct
B. Incorrect – This concern is associated with using the RWCU system in letdown or recirc.
C. Incorrect – this method is not permitted without establishment of primary and secondary containment.
D. Incorrect – this method is not permitted without establishment of primary and secondary containment.

Per ON-149-001, Rev.19, page 19

When using CS method, RHR in suppression pool Cooling Mode is used to provide cooling as required. Preferred SRV's for this evolution are designated and are those having discharge piping with no bends or long runs of nearly horizontal piping. Although these SRV's are preferred, if not available, any other SRV may be used. One concern in this configuration is that rapid cooling due to high flow of cooler water from suppression pool may result in exceeding 90°F/hour cooldown rate.

In Mode 3 with Primary and Secondary containment established, reactor pressure may be maintained greater than 20 psig (to clear column of water from WV discharge downcomers if steam flow is routed or diverted there) and less than 98 psig (below reactor high pressure isolation). Steam may be routed to the main condenser or suppression pool and methods of makeup previously discussed may be used to maintain level. This method may also be used in Mode 4 with Primary and Secondary containments established. Reactor pressure and temperature may be allowed to rise until within pressure limits cooling by boiling as above. One factor to be considered in Mode 4 is that using this method will result in entering the Emergency Plan. The major factor that must be determined before using this method is that Primary and Secondary containments are established in order to prevent release of radioactive materials to the environment.

Technical Reference(s): ON-149-001, Rev. 20, page 21/22 discussion (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 X

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #	2.4.10	
Importance Rating		3.1

Knowledge of annunciator response procedures.

Proposed Question: SRO 100

Unit 1 is at 100 percent power.

Annunciator alarm AR-107-D04 ARI DIV 1 INOP/BYPASS was received. Electrical Maintenance investigated cause of the alarm and reports a loss of power to the ARI DIV I logic.

What is the impact of this failure and what action, if any, is required?

A. ATWS-ARI trip input signals to Division 1 RPS logic are inoperable. Place channel in trip condition within One (1) hour.

B. Manual and automatic actuation of ARI are inoperable. Restore ATWS-ARI trip capability within 14 days.

C. ATWS-ARI trip input signals to Division 1 RPS logic are inoperable. Place channel in trip condition within 12 hours.

D. Division II manual and automatic actuation of ARI are operable. No further action is required.

Proposed Answer: B

Explanation (Optional):

A	Incorrect. A separate 125 VDC source provides power to Div 1 backup scram valves, therefore, the function remains operable.
B	Correct - Both divisions must energize to cause scram air header isolation and venting. Since a power loss is involved neither manual nor automatic actuation is operable. TRO 3.1.1 requires trip capability restored within 14 days.
C	Incorrect. Div 2 ATWS-ARI remains operable, however, both divisions must energize to cause scram air header isolation and venting. Rods scram times are not extended to 25 seconds in this condition.
D	Incorrect. ATWS-ARI does not provide trip signals to RPS, it is an independent function to RPS.

Per TRM 3.1.1

Per TM-OP-058-ST, Rev. 3, pages 45 and 46.

ALTERNATE ROD INSERTION (ARI) TRIP SYSTEM

ARI is a fully independent system from RPS, though the function caused by each system is similar.

ARI Logic has two divisions. Each division controls the solenoid on one Vent valve and one Block Valve. Both divisions must actuate to block and vent the Scram Air Header. The ARI Logic is said to be "two-out-of-two-twice" logic.

Technical Reference(s): TRM3.1.1, ____ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____ (As available)

Question Source:	Bank #	TMOP055/12727	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	X	