Susquehanna ILO NRC Exam - Question Repetition Analysis

Region I examiners verified that the 2003 ILO exam to be administered in August 2003 did not contain an inappropriate number of questions repeated from the licensee's two previous ILO exams (administered in August 2002 and August 2001). The methodology used for the verification was based on an analysis of each exam's K/A content. Individual K/A lists for each of the three exams were prepared and a comparison made between the lists. This analysis revealed that the 2003 exam repeated 9 K/As from the 2002 exam and 7 K/As from the 2001 exam; two of these K/As were common to both the 2001 and 2002 exams, indicating that the 2003 exam used a total of 14 K/As that had been used in the two previous exams. The examiners checked the individual questions developed from those 14 K/As and verified that in all 14 cases entirely new questions had been developed by the licensee for the 2003 exam (i.e., the 2003 exam repeats zero questions from the previous two exams administered at the facility).

WRITTEN REFERENCE

- EOP FLOWCHARTS – NO ENTRY CONDITIONS
- ON-145-004 "RPV WATER LEVEL ANOMALY"
- ON-102-620 "LOSS OF 125V DC BUS 1D620"
- STEAM TABLES

Candidate References LOC 19 NRC Exam

Reference	RO	SRO	Question #
Emergency Operating Procedures (No entry conditions)	X	X	13, 79, 80, 83, 86,
Technical Specifications 3.3 Instrumentation TS 3.3.6 Primary Containment Isolation Instrumentation		X	89,
Technical Requirements 3.5 Emergency Core Cooling and RCIC		X	90
Technical Specifications 3.4 Reactor Coolant System TS 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - - Hot Shutdown TS 3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System - - Cold Shutdown	X	X	92
Technical Specifications 3.5 Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System 3.5.1 ECCS - Operating		X	29 RO question
Technical Specifications 3.6 Containment Systems TS 3.6.1.3 Primary Containment Isolation Valves TS 3.6.2.1 Suppression Pool Average Temperature		X	89, 95
NDAP-QA-0720 Station Report Matrix and Reportability Evaluation Guidance Attachment A Attachment E Attachment F Attachment G Attachment H Attachment I Attachment K Attachment M Attachment Q Attachment T		X	85 (Att G), 99 (Att Q & T)

PP&L STANDARD EXAM SHEET

Course #: PC017 Course: LOC 19 NRC Operator EXAM

First Name: ____________ Last Name: ____________ Employee Number: ______

WRITTEN EXAM ANSWER KEY - SUSQ

Social Security Number

Test Form RO/SRO

A B C D

Test Date: __/__/__

Test Series: ______

Test Number: ______

Test Taking is an Individual Effort: Any test misconduct is a violation of the Academic Honesty Policy (NTP-QA-14.2) and the PP&L Standards of Conduct and Integrity.

Signature: ____________________

______ # Correct ______ % Score

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

SSES LOC 19 NRC Exam

1 Unit 1 startup was in progress with Reactor Power at 74% when RECIRC PUMP A DSCH HV-143-F031A open position logic failed causing a run-back. The limit switch logic problem has been repaired and the runback logic is to be reset.

Which of the following actions must be completed to reset the runback logic?

A. Lower GEN 1A & 1B SPEED control signal to slightly lower Recirc Pump speed, reset Limiter #1, monitor for speed change.

B. Lower GEN 1A SPEED control signal to slightly lower Recirc Pump speed, reset Limiter #2, monitor for speed change.

C. Lower GEN 1A SPEED control signal to slightly lower Recirc Pump speed, reset Limiter #1 & #2, monitor for speed change.

D. Lower GEN 1A SPEED control signal to slightly lower Recirc Pump speed, reset Limiter #1, monitor for speed change.

Question Data

Answer: D lower GEN 1A SPEED control signal to slightly lower Recirc pump speed, reset limiter #1, monitor for speed change.

Explanation/Justification:

A. Runback does not affect both pumps, only 'A'

B. #2 limiter is caused by Low Reactor Level or Circ water pump trip

C. Valve position does not feed both limiters.

D. Correct answer

Sys #	System	Category	KA Statement
295001	**Partial or Complete Loss of Forced Core Flow Circulation**	**Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:**	**Recirculation flow control system**

K/A# **295001.AA1.05** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-164-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

2 Unit 1 & 2 are operating at 100% power with a normal electrical lineup when a loss of Startup Bus 10 occurs.

Assuming no operator actions, and all equipment functions as designed. What will be the status of the Unit 1 'A' ESS Bus Breakers following this event?

A. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Closed
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Closed

B. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

C. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Closed
D/G A to Bus 1A - (1A20104) Breaker Open

D. Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Closed
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Open
D/G A to Bus 1A - (1A20104) Breaker Open

Question Data

Answer: C Transformer 101 (0X201) to Bus 1A - (1A20101) Breaker Open
Transformer 201 (0X203) to Bus 1A - (1A20109) Breaker Closed
D/G A to Bus 1A - (1A20104) Breaker Open

Explanation/Justification:

A. No power available through 01 breaker and 04 breaker would not close if 01 breaker still closed on bus.
B. The D/G output breaker would be expected to be closed if both the 01 and 09 breakers open.
C. Correct answer
D. No power available through 01 breaker

Sys #	System	Category	KA Statement
295003	**Partial or Complete Loss of A.C. Power**	**Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following:**	**Emergency generators**

K/A# 295003.AK2.02 K/A Importance 4.1 Exam Level RO

References provided to Candidate **None** Technical References: **ON-003-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

3 125 VDC power from 1D614 to the Auto Depressurization System (ADS) logic is lost.

What happens if a valid initiation signal is then received with the loss of power?

A. No ADS Valves will open.

B. ADS logic "B" will still initiate ADS; only 3 ADS Valves will open.

C. ADS logic "A" will receive backup power from 125 VDC 1D624; all ADS Valves will open.

D. ADS logic "B" will still initiate ADS; all ADS Valves will open.

Question Data

Answer: D ADS logic "B" will still initiate ADS, All ADS valves will open

Explanation/Justification:

A. All ADS valves will open.
B. 6 ADS valves will open
C. 1D614 does not have an alternate
D. correct answer, Either division of ADS will provide actuation of all 6 ADS valves

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Sys #	System	Category	KA Statement
295004	**Partial or Complete Loss of D.C. Power**	**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**

K/A# **295004.AK1.03** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-083**

Question Source: **Modified** **Fitzpatrick, 1992** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

4 The following alarm is received for Division II:

TURB STOP VLV CLOSURE TRIP AR-104-E02

Which of the following lists the items causing the alarm?

A. Turbine tripped.
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

B. Turbine tripped.
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel A.

C. Turbine reset and speed not selected.
Stop Valves or Control Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

D. Turbine reset and speed not selected.
Stop Valves 1 and 3 or 2 and 4 < 98% open.
Loss of power to RPS Channel B.

Question Data

Answer: A Turbine tripped.
Stop Valves 1 and 3 or 2 and 4 < 95% open.
Loss of power to RPS Channel B.

Explanation/Justification:

A. Correct answer
B. Loss of power from B RPS causes Div 2 alarm.
C. Control valves will not cause alarm.
D. Valve position is <95%

Sys #	System	Category	KA Statement
295005	**Main Turbine Generator Trip**	**Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:**	**Turbine valve position**

K/A# 295005.AA2.03 K/A Importance 3.1 Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-104-001**

Question Source: **New** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

5 The following post scram conditions exist:

Automatic scram was initiated on low RPV level
Both Recirc Pumps are tripped
'A' Recirc loop temperature 500 deg F
'B' Recirc loop temperature 480 deg F
RPV level is 20 inches and rising slowly
RPV pressure is 950 psig and stable
All rods are fully inserted
RWCU is isolated
Calculated steam dome to bottom head drain differential temperature is 110 deg F

Which one of the following identifies the action to be taken due to vessel thermal stratification per ON-100-101, Scram?

A. Raise level to promote natural circulation; restart 'A' Recirculation Pump.

B. Raise level to promote natural circulation; restart 'B' Recirculation Pump.

C. Do not raise level to promote natural circulation; do not restart a Recirculation Pump.

D. Do not raise level to promote natural circulation; restart both Recirculation Pumps.

Question Data

Answer: A Raise level to promote natural circulation, restart 'A' Recirculation pump.

Explanation/Justification:

A. Candidate needs to know with <145 deg F Delta between dome and bottom head temperature, water level is raised to promote natural circ. Candidate needs to know to restart a Recirc pump the Recirc loop temperature must be within 50 degrees of the reactor vessel temperature. For the case given, 950 psig Rx pressure is 540 deg F from the steam table minus the loop temperatures is >50 deg F delta preventing a start of the 'B' recirculation pump.

B. Can not restart a Recirc pump

C. Level may be raised

D. Level may be raised, Recirc can not be restarted

Sys #	System	Category	KA Statement
295006	**SCRAM**	**Ability to operate and/or monitor the following as they apply to SCRAM:**	**Recirculation system**

K/A# 295006.AA1.04 K/A Importance 3.1 Exam Level RO

References provided to Candidate **Steam Tables, TS 3.4.10** Technical References: **ON-100-101**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **10CFR55.41.10**

6 Given the following conditions:

- A Unit 1 fire has resulted in the closure of all Outboard Main Steam Isolation Valves from 100 percent power
- The Immediate Operator Actions of ON-100-009, "Control Room Evacuation" were completed
- High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) both automatically initiated and are injecting
- All Remote Shutdown Panel (RSP) Control Transfer Switches have been placed in **EMERG** position.
- The PCO trips RCIC when reactor water level rises to +54 inches
- All automatic functions occur as designed

If no operator actions are taken, how will HPCI and RCIC function to control Reactor water level?

A. HPCI will re-initiate automatically when water level reaches -38 inches.

B. RCIC will re-initiate automatically when water level reaches -30 inches.

C. HPCI and RCIC will re-initiate automatically when water level drops to -38 inches.

D. No automatic actions will occur. Operator must start HPCI or RCIC for level control

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Question Data

Answer: A HPCI re-initiate automatically when water level reaches -38 inches.

Explanation/Justification:

A. correct answer, There are no HPCI controls at the Remote Shutdown Panel thus no impact on the HPCI system when all the transfer switches are re-aligned. HPCI will trip at +54 inches as it will based on the conditions provided in the question. With no operator actions being taken water level will lower to the point of the HPCI low level initiation and HPCI will start and feed water level to the high level trip point of +54 inches.

B. RCIC does not have auto initiation capability when the Remote Shutdown Panel switches are transferred and will not automatically start when level drops to -30 inches.

C. RCIC does not have auto initiation capability when the Remote Shutdown Panel switches are transferred and will not automatically start when level drops to -30 inches.

D. HPCI will automatically start when level lowers to -38 inches.

Sys #	System	Category	KA Statement
295016	**Control Room Abandonment**	**Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:**	**Reactor water level**

K/A# **295016.AA1.06** K/A Importance **4.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-100-009**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

7 Given the following conditions:

- Unit 2 is operating at 100% power
- The "2B" Reactor Protection System (RPS) Bus is on the alternate power supply
- The "2A" RPS MG Set has just tripped

Unit 2 may operate in Mode 1 for a limited amount of time based upon:

A. the length of time for restoration of Emergency Switchgear cooling.

B. the availability of the Reactor Building Sump Pumps.

C. the Containment instrument gas supply to the Inboard MSIVs.

D. the availability of the Reactor Recirculation Pumps.

Question Data

Answer: D the availability of the Reactor Recirculation Pumps.

Explanation/Justification:

A. cooling will be restored if 'A' equipment in service or no effect if 'B' equipment in service.

B. continued operation based on being able to complete leakage surveillance, not an immediate concern

C. not an immediate concern

D. correct answer, loss of RPS causes containment isolation and loss of cooling water to the Recirc pumps

Sys #	System	Category	KA Statement
295018	**Partial or Total Loss of CCW**	**GENERIC**	**Knowledge of abnormal condition procedures**

K/A# **295018.2.4.11** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-158-001**

Question Source: **Modified** **SSES, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

8 With Unit 1 at full power, Annunciator AR-124-B01, INSTRUMENT AIR HEADER LO PRESSURE alarms.

What is this alarm actually monitoring and what is expected to happen if the condition continues to degrade?

A. Scram air header pressure is less than 65 psig. If not corrected, Scram Discharge Volume Vent and Drain Valves will fail closed preventing a scram.

B. Scram air header pressure is less than 75 psig. If not corrected, the scram valves will begin to open, scramming in rods.

C. Instrument air header pressure is less than 80 psig. If not corrected, the scram valves will begin to open, drifting in rods.

D. Instrument air header pressure is less than 80 psig. If not corrected, Scram Discharge Volume Vent and Drain Valves will fail closed preventing a scram.

Question Data

Answer: C Instrument air header pressure is less than 80 psig. If not corrected, the scram valves will begin to open, drifting in rods.

Explanation/Justification:

- **A.** Alarm is measuring IA header pressure SDV vent & drain valves closing will not prevent a scram.
- **B.** Alarm is measuring instrument air header pressure.
- **C.** correct answer ON-118-001 discussion section: scram inlet and outlet valves will begin drifting open causing random control rod insertion.
- **D.** SDV vent & drain valves closing will not prevent a scram.

Sys #	System	Category	KA Statement
295019	**Partial or Total Loss of Inst. Air**	**GENERIC**	**Ability to verify that the alarms are consistent with the plant conditions**

K/A# **295019.2.4.46** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-118-001**

Question Source: **Modified** **Quad-Cities 1 & 2, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

SSES LOC 19 NRC Exam

9 Why is the operator directed to ensure Reactor water level greater than +45 inches when RHR Shutdown Cooling is lost?

A. To ensure natural circulation occurs since forced circulation is lost.

B. To ensure the feedwater spargers effectively mix the feedwater added to the Reactor Vessel.

C. To maintain Reactor Narrow range water level instrumentation on scale when the Reactor water temperature increases.

D. To ensure the water from the annulus area of the Reactor flows into the core shroud area as level decreases inside the shroud.

Question Data

Answer: A to ensure natural circulation occurs since forced circulation is lost.

Explanation/Justification:

A. correct answer, ON-149-001 discussion, >+45 raises water level above steam separator to establish natural circulation

B. Feedwater sparger mixing is a concern during ATWS not loss of S/D cooling. Any feedwater addition has no affect on why level is maintained >+45.

C. Narrow and wide range upscale <200 psig in vessel with actual level >+45

D. with level >+45 there is no difference in water level inside or outside of shroud.

Sys #	System	Category	KA Statement
295021	**Loss of Shutdown Cooling**	**Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING:**	**Establishing alternate heat removal flow paths**

K/A# **295021.AK3.05** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-149-001**

Question Source: **Modified** **Monticello 1, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

10 Given the following conditions:

- A Railroad Access Shaft Exhaust Duct Radiation High signal is received.
- Standby Gas Treatment responds as required.
- All Zone I differential pressures are -.27 inches WG.

Select the Standby Gas Treatment system response to Zone II differential pressure decreasing to -.23 inches WG.

A. Outside Cooling air dampers will modulate open.

B. Outside makeup air dampers will modulate closed.

C. Standby Gas Treatment dampers will not adjust for Zone II delta P.

D. Standby gas treatment fan inlet vanes will modulate open.

Question Data

Answer: C *Standby Gas Treatment dampers will not adjust for Zone II delta P.*

Explanation/Justification:

A. Outside cooling air dampers modulate based on gas stream temperature., not any Z-I, II or III delta pressure.
B. Outside makeup air dampers modulate based on suction pressure, not Z-I, II or III delta pressure.
C. correct answer, relay circuit excludes the signal from a zone that does not have an isolation signal present
D. Inlet vanes vary position to maintain flow rate

Sys #	System	Category	KA Statement
295023	**Refueling Accidents**	**Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:**	**Standby gas treatment/FRVS**

K/A# **295023.AA1.07** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-070**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

11 Given the following parameters:

- Drywell pressure	3.5 psig and rising
- Drywell temperature	145 degrees F and rising
- Suppression Chamber pressure	4.6 psig and rising
- Suppression Pool water temperature	87 degrees F and steady

What event from below would cause the conditions listed above?

A. A safety relief valve tail pipe has broken in the Suppression Chamber while the valve is open.

B. A pipe break into the drywell has occurred with a Suppression Chamber to Drywell vacuum breaker open.

C. A downcomer vacuum breaker has failed open during a recirculation leak to the Drywell.

D. A recirculation line partial break has occurred with all containment parameters responding as designed.

Question Data

Answer: A A safety relief valve tail pipe has broken in the Suppression Chamber while the valve is open.

Explanation/Justification:

A. correct answer, energy into chamber but not into pool, vacuum breakers opening back to drywell when d/p high enough

B. downcomer vacuum breakers are designed to be open for these conditions, equalize pressure across the drywell floor when drywell pressure less than chamber pressure

C. only one vacuum breaker failing would not provide a vent path to Suppression pool atmosphere.

D. all parameters way too low, especially pool temperature

Sys #	System	Category	KA Statement
295024	**High Drywell Pressure**	**Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:**	**Suppression chamber pressure: Plant-Specific**

K/A# **295024.EA2.04** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-059**

Question Source: **Modified** **Susquehanna, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

12 The reason for maintaining Suppression Pool temperature below the Heat Capacity Temperature Limit curve in EOP-100-103, PRIMARY CONTAINMENT CONTROL is:

A. To assure primary containment vent valve opening capability following RPV depressurization.

B. To provide adequate subcooling in the pool to prevent chugging of the SRV downcomers.

C. To assure the containment design pressure will not be exceeded due to compression of the non-condensable gases.

D. To assure the Suppression Pool can vent non-condensables back to the Drywell vapor space.

Question Data

Answer: A to assure primary containment vent valve opening capability following RPV depressurization.

Explanation/Justification:

A. correct answer

B. suppression chamber pressure of 13 psig pertains to chugging of SRVs

C. assumption that all non-condensables are in the suppression chamber, reason for shape of curve, not reason for curve

D. the curve assumes that all the non-condensibles are in the suppression chamber and not vented back to the drywell.

Sys #	System	Category	KA Statement
295025	**High Reactor Pressure**	**GENERIC**	**Knowledge of the specific bases for EOPs**

K/A# 295025.2.4.18 K/A Importance 2.7 Exam Level RO

References provided to Candidate **None** Technical References: **EO-000-103**

Question Source: **Modified** **Clinton 1, 2000** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

13 Given the following conditions:

- Unit 1 is in an ATWS condition
- Reactor pressure is 950 psig
- Suppression Pool temperature is 175 degrees F

Based on the listed conditions, if Suppression Pool level began to decrease, when would an emergency blowdown be REQUIRED after the rods are inserted?

A. 15.5 feet

B. 17 feet

C. 19.5 feet

D. 22 feet

Question Data

Answer: A 15.5 feet

Explanation/Justification:

A. Correct answer, using HCTL curve.
B. at 17 feet go to RPV control
C. 19.5 is RCIC room equalization value.
D. minimum normal level of the suppression pool.

Sys #	System	Category	KA Statement
295026	**Suppression Pool High Water Temperature**	**Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:**	**Suppression pool level**

K/A# 295026.EA2.02 K/A Importance 3.8 Exam Level **RO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-000-103**

Question Source: **Modified** **Peach Bottom, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

14 What would be the consequence of NOT limiting Drywell Spray flow for the first 30 seconds of operation?

NOT limiting Drywell Spray flow for the first 30 seconds of operation will:

A. cause the Drywell Spray Initiation Limit (DWSIL) of 13 psig to be exceeded.

B. cause a pressure drop fast enough to exceed the Containment design differential pressure.

C. cause rapid pressure drop with minimal Suppression Chamber vapor to support Suppression Pool - Drywell vacuum breaker operation.

D. cause excessive thermal and mechanical shock to the Drywell downcomers.

Question Data

Answer: B cause a pressure drop fast enough to exceed the Containment design differential pressure.

Explanation/Justification:

A. limiting flow allows initiation of sprays at any temperature (not pressure), or without concern in all regions of DWSIL curve.

B. Correct answer.

C. Vapor in drywell is issue not suppression pool

D. not a consideration

Sys #	System	Category	KA Statement
295028	**High Drywell Temperature**	**Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE:**	**Drywell spray: Mark-I&II**

K/A# **295028.EA1.01** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-103**

Question Source: **Modified** **OYSTER CREEK, 1991** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

15 The plant is in a LOCA with ECCS Systems injecting to the reactor.

Suppression Pool level has dropped to 20.5 feet.

Which of the following is a condition that exists due to Suppression Pool level?

A. Indicated Suppression Pool average temperature is not valid.

B. The SRV tailpipe T-Quenchers have been uncovered.

C. The HPCI Turbine Exhaust has been uncovered.

D. Indicated Containment Pressure is not valid.

Question Data

Answer: A Suppression Pool average temperature is not valid.

Explanation/Justification:

A. Correct answer, must use lower SPOTMOS sensors.
B. Not applicable, SRV tailpipe exhausts uncover at 5'
C. HPCI Turbine exhaust is a concern at <17'
D. Containment pressure indication is available.

Sys #	System	Category	KA Statement
295030	**Low Suppression Pool Water Level**	**Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:**	**Suppression pool temperature**

K/A# **295030.EA2.02** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-103**

Question Source: **Modified** **Grand Gulf 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

16 Unit 2 is in Mode 4 with the following conditions:

'A' RHR Pump running in Shutdown Cooling mode.
Both Recirculation MG Sets out of service
RWCU isolated
CRD Pumps shut down
Condensate and Feedwater out of service
A TOTAL loss of Shutdown Cooling occurs with vessel water level at +40 inches.

The Off-Normal for loss of shutdown cooling directs Reactor Vessel level to be raised to + 90 to 100 inches.

What will be the consequences of not raising the water level for a period of 6 hours?

A. Indicated Reactor Pressure will increase.
Indicated Bottom Head temperature will increase.
Recirc Loop temperatures will increase.
Reactor vessel metal temperature at beltline will increase.

B. Indicated Reactor Pressure will increase.
Indicated Bottom Head temperature remains constant.
Recirc Loop temperatures remains constant.
Reactor vessel metal temperature at beltline will increase.

C. Indicated Reactor Pressure will remain constant.
Indicated Bottom Head temperature will increase.
Recirc Loop temperatures remains constant.
CRD mechanism High temperature will alarm.

D. Indicated Reactor Pressure remains constant.
Indicated Bottom Head temperature remains constant.
Recirc Loop temperatures remains constant.
Reactor vessel metal temperature at vessel head will increase.

Question Data

Answer: B Indicated Reactor Pressure will increase
Indicated Bottom Head temperature remain constant.
Recirc Loop temperatures constant
Reactor vessel metal temperature at beltline increase

Explanation/Justification:

A. Bottom head temperature will not increase since there is no flow in the region with CRD, Recirc and RWCU out of service. There is no flow in the recirc loop so temperature will not increase in the recirc loop.

B. correct answer, indicated reactor pressure will increase due to decay heat raising the temperature of the coolant in the immediate core area. Bottom head temperature remains constant since there is no flow in and around the bottom head with RWCU and CRD out of service. Recirc loop temperatures will not change since there is no natural circulation out side of the core barrel due to no flow path through the steam separators. Vessel beltline temperature will rise due to radiant and conductive heating from the core region.

C. Pressure will increase due to heatup of water in the core region. Bottom head temperature will not increase since there is no flow in the region with CRD, Recirc and RWCU out of service. CRD mechanisms will not alarm since there is no hot water to heat up the drives.

D. Pressure will increase due to heatup of water in the core region.

Sys #	System	Category	KA Statement

SSES LOC 19 NRC Exam

295031	Reactor Low Water Level	Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL:	Natural circulation: Plant-Specific

K/A# 295031.EK1.02 K/A Importance 3.8

Exam Level RO

References provided to Candidate None

Technical References: ON-249-001/SOER 8202

Question Source: New

Level Of Difficulty: (1-5) 3

Question Cognitive Level: Fundamental

10 CFR Part 55 Content: 55.41

17 Following a LOCA on Unit 1 the following conditions exist:

- All control rods have inserted.
- RPV water level is -165 inches (actual level) and slowly lowering.
- RPV pressure is 815 psig and stable.
- No offsite power is available, no Diesel Generators are running.
- HPCI and RCIC are not available.
- The Diesel Fire Pump is not running.

Which of the following will maintain STEAM COOLING?

A. The reactor must be rapidly depressurized per EO-100-112.

B. RPV water level remains between -161 and -205 inches.

C. RPV pressure remains between 815 psig and 1087 psig.

D. Steam cooling is not possible under these conditions.

Question Data

Answer: B RPV water level remains between -161 and -205 inches.

Explanation/Justification:

A. Rapid depressurization accelerates the rate of inventory loss and pressure drop, thereby shortening the time that steam cooling is maintained.

B. correct answer, Steam cooling occurs when water heated in the core boils turns to steam and rises in the bundles cooling the upper portions. This occurs below -161 inches and continues to -205 inches (MZIRWL). For this to occur there must be a steam flow path and zero injection. From the stated conditions the break is providing a flow path (pressure is NOT rising).

C. RPV pressure must be stable or decreasing. Raising reactor pressure will be detrimental to the steam cooling. If RPV pressure is rising, the assumptions of the MZIWL calculation are no longer valid and the core may not be adequately cooled.

D. The given conditions do support steam cooling.

Sys #	System	Category	KA Statement
295031	**Reactor Low Water Level**	**Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL:**	**Adequate core cooling**

K/A# **295031.EA2.04** K/A Importance **4.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-102**

Question Source: **Modified** **Susquehanna, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

SSES LOC 19 NRC Exam

18 Under certain ATWS conditions, the EOPs direct the operators to take action to deliberately lower RPV level in order to reduce reactor power.

Which of the following describes why reactor power decreases as RPV level is lowered?

Negative reactivity is added from increased void fraction due to:

A. reducing the natural circulation flow which increases the rate of steam removal.

B. uncovering the feedwater spargers to reduce core inlet subcooling.

C. increasing the amount of carryunder through the steam dryer and separators.

D. reducing NPSH available to the Jet Pumps, reducing flow the pumps deliver.

Question Data

Answer: B uncovers the feedwater spargers to reduce core inlet subcooling.

Explanation/Justification:

A. steaming rate decreases due to increased core inlet temperature and reduced moderation.
B. Correct answer
C. steaming rate decreases as power decreases, total carryover decreases.
D. quality is a measure of steam to moisture, not applicable to core inlet parameters.

Sys #	System	Category	KA Statement
295037	**SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown**	**Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:**	**Reactor water level**

K/A# **295037.EK2.09** K/A Importance **4.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-100-113**

Question Source: **Modified** **WPPSS 2, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

19 EO-100-105, "Radioactivity Release Control", directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building, except those systems required to support EOP/DSP actions.

These systems are specifically exempted from isolation, because:

A. isolation of a EOP support system requires an upgrade of the Emergency Classification.

B. they are required to support alternate reactor depressurization methods.

C. additional radiological consequences from them are unlikely.

D. isolation may ultimately result in a much larger uncontrolled radiological release.

Question Data

Answer: D isolation may ultimately result in a much larger uncontrolled radiological release.

Explanation/Justification:

A. not a consideration for these conditions
B. not true
C. alternate depress methods are part of EOPs
D. correct answer

Sys #	System	Category	KA Statement
295038	**High Off-Site Release Rate**	**Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE:**	**System isolations**

K/A# **295038.EK3.02** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-105**

Question Source: **Modified** **Susquehanna, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

20 The T-20 Transformer developed an internal fault causing a fire and automatic actuation of the Fire Protection Deluge System. The Fire Protection System functions as designed. For these conditions the Diesel Driven Fire Pump will automatically start if the fire protection header pressure drops to:

A. 85 psig.

B. 95 psig

C. 105 psig

D. 125 psig

Question Data
Answer: A 85 psig.

Explanation/Justification:
A. correct answer,
B. The auto start of the Motor Driven Fire pump
C. The auto start of the Jockey Fire Pump
D. The auto shutdown of the Jockey Fire Pump

Sys #	System	Category	KA Statement
600000	**Plant Fire On Site**	**GENERIC**	**Ability to locate control room switches / controls and indications and to determine that they are correctly reflecting the desired plant lineup**

K/A# **600000.2.1.31** K/A Importance **4.2** Exam Level **RO**
References provided to Candidate **None** Technical References: **TM-OP-013**
Question Source: **New** Level Of Difficulty: (1-5) **2**
Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

21 Unit 2 is starting up with power at 50%.

The third Reactor Feed Pump has just been placed in service.

A Reactor Recirc system runback to 30% has occurred. The runback is due to the failure of the Feedwater Level Control System total feed flow signal.

What is the expected plant response with no operator actions?

A. Reactor Recirc Pump Trip signal

B. High Water RPV level alarm

C. Feedwater level control system Setpoint Setdown signal

D. Low Reactor Water Level Alarm

Question Data

Answer: B High Water RPV level alarm

Explanation/Justification:

A. No trip signal to Recirc pumps on high level.

B. Correct answer, Recirc runback caused by failure of feedwater flow signal to less than 20% flow, FWLC system sees a steam feed mismatch with feed flow low as compared to steam flow, FWLC will increase feed flow to cancel Steam/Feed mismatch UNTIL level signal error over-rides steam feed mismatch thus, vessel level will increase.

C. Level will fail high not low.

D. Level will fail high not low.

Sys #	System	Category	KA Statement
295008	**High Reactor Water Level**	**Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL:**	**Feed flow/steam flow mismatch**

K/A# **295008.AK1.03** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-045**

Question Source: **Modified** **Lasalle 1, 1996** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

22 Unit 1 is at 100% power in a normal lineup when Annunciator RX WATER HI-LO LEVEL AR-101-B17 actuates.

The following indications are observed at 1C651:

- The Master Feed Water Level Controller LIC-C32-1R600 is in AUTO and set at 34.
- 1'A' Reactor Feed Pump Speed Controller SIC-C32-1R601A demand signal is slowly lowering.
- 1'B' Reactor Feed Pump Speed Controller SIC-C32-1R601B demand signal is slowly rising.
- 1'C' Reactor Feed Pump Speed Controller SIC-C32-1R601C demand signal is slowly rising.
- The failed RFP A Speed Controller is placed in MANUAL.

Which of the following is the operator response for this situation?

A. Adjust feed flow to equal B & C Reactor Feed Pumps, or lower Motor Speed Changer, activate Hydraulic Jack; control speed to control flow.

B. Adjust feed flow to equal B & C Reactor Feed Pumps, or activate Hydraulic Jack, lower Motor Speed Changer; control speed to control flow.

C. Adjust Master Feed Water Level Controller to maintain RPV water level ~35"; dedicate operator for RPV level.

D. Adjust feed flow to equal B & C Reactor Feed Pumps, or activate Hydraulic Jack, lower Electrical Automatic Positioner (EAP); control speed to control flow.

Question Data

Answer: A Adjust feed flow to equal B & C Reactor Feedpumps, or, lower Motor Speed Changer, activate Hydraulic Jack, control speed to control flow.

Explanation/Justification:

A. correct answer,
B. Lower MSC first then set Hydraulic Jack.
C. 'A' RFP controller has failed, attempting to control level with master controller will not work.
D. Lower MSC first then set Hydraulic Jack.

Sys #	System	Category	KA Statement
295009	**Low Reactor Water Level**	**Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL:**	**Reactor water level control**

K/A# **295009.AA1.02** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-145-001**

Question Source: **Modified** **Duane Arnold 1, 1999** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

23 A Reactor scram resulted in water level dropping to -31 inches on Wide Range Level.

- Reactor level has since recovered to + 10 inches on Narrow Range.
- Reactor Pressure is being maintained with the Turbine Bypass Valves.
- The maximum Reactor Pressure during the transient was 1080 psig.
- The operators have placed the 'A' RHR Pump in Suppression Pool Cooling.
- Suppression Pool Water Temperature is rising.

Which of the following is the cause of rising Suppression Pool water temperature?

A. RCIC in operation, F048A HX A SHELL SIDE BYPS open; F047A RHR HX A SHELL SIDE INLET closed.

B. RCIC and HPCI in operation, F048A HX A SHELL SIDE BYPS open; F047A RHR HX A SHELL SIDE INLET closed.

C. HPCI in operation, F048A HX A SHELL SIDE BYPS open; F047A RHR HX A SHELL SIDE INLET open.

D. SRV actuation, F048A HX A SHELL SIDE BYPS open; F047A RHR HX A SHELL SIDE INLET closed.

Question Data

Answer: A RCIC in operation, F048A HX A SHELL SIDE BYPS open, F047A RHR HX A SHELL SIDE INLET closed.

Explanation/Justification:

A. correct answer, some cooling being provided with some flow forced through heat exchanger, heat input from RCIC auto start at -30", level not low enough to auto start HPCI, pressure not high enough to open SRV.

B. HPCI did not receive a auto start signal.

C. *Pressure did not get high enough to open SRV.*

D. Pressure did not get high enough to open SRV.

Sys #	System	Category	KA Statement
295013	**High Suppression Pool Temperature**	**Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:**	**Suppression pool cooling**

K/A# **295013.AA1.01** K/A Importance **3.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-149-005**

Question Source: **Modified** **Grand Gulf 1, 2000** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

24 A scram condition has occurred on Unit 2 with the following conditions present:

Scram solenoids de-energized
Backup scram valves energized
ARI initiated (by operator)
Mode switch in SHUTDOWN
30% of the control rods DID NOT insert

Which ONE (1) of the following would cause the 30% failure to scram?

A. Scram Discharge Volume is full.

B. Failure of the scram air header to depressurize.

C. Failure of the RPS scram logic.

D. Scram Discharge Volume (SDV) Vent and Drain Valves failing to close.

Question Data

Answer: A Scram Discharge Volume is full.

Explanation/Justification:

A. Correct answer, Scram solenoids de-energized vent the scram air header venting air from the scram valves allowing rod motion. If there were an air blockage preventing venting the scram air header indication would be that none of the rods moved. Backup scram valves energized to operate and would also vent the scram air header. The indication provided is 30% of the rods did not fully insert indicating a hydraulic lock on the scram discharge volume.

B. If the air header did not depressurize it would be expected that none of the rods would move.

C. If there were a failure of the RPS scram logic it would be expected that none of the rods would move.

D. Scram discharge vent and drain valves failing open would not inhibit a scram. Failure of the valves will cause a radioactive release but not prevent a scram.

Sys #	System	Category	KA Statement
295015	**Incomplete SCRAM**	**Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM:**	**CRD hydraulics**

K/A# **295015.AA1.01** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-055**

Question Source: **NRC Exam** **Browns Ferry 2, 9/17/2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

25 The following statement exists in EO-100-105, RADIOACTIVITY RELEASE CONTROL:

" IF TURBINE OR RW BUILDING HVAC NOT IN SERVICE RESTART APPLICABLE HVAC AS REQ'D"

Which of the following is the basis for keeping the Turbine Building Ventilation System in operation while executing EO-100-105, RADIOACTIVITY RELEASE CONTROL?

Having Turbine Building Ventilation in service:

A. maintains Turbine Building pressure above Reactor Building Pressure.

B. prevents having an unmonitored ground release from the Turbine Building.

C. prevents a reactor scram due to high temperature in the MSL tunnel.

D. ensures adequate dilution of the gases discharged through the Turbine Building Vent.

Question Data

Answer: B *prevents having an unmonitored ground release from the Turbine Building.*

Explanation/Justification:

A. Turbine Building pressure has no correlation to Reactor Building pressure
B. correct answer, operation with no ventilation in service will lead to an unmonitored ground level release
C. the EOP is not addressing high temperature in the steam tunnel.
D. the EOP is not addressing dilution of releases from the ventilation system

Sys #	System	Category	KA Statement
295017	**High Off-Site Release Rate**	**Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE:**	**Protection of the general public**

K/A# **295017.AK1.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-105**

Question Source: **Modified** **Nine Mile Point 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

26 Suppression Pool Cleanup is in operation discharging to Radwaste to lower the level of the Unit 1 Suppression Pool.

An inadvertent trip of the Division I logic for High Drywell Pressure is actuated and confirmed by alarms received in the Control Room.

What would be the trend of the Rad Waste Collection Tank level indicator as a result of the High Drywell Pressure?

A. LRW Collection Tank Level trend is rising, Suppression Pool Level is lowering, Suppression Pool Cleanup Pump tripped, and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 and OB ISO HV-15768 are open.

B. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup Pump tripped, and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 is closed.

C. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup Pump running, and SUPP POOL WTR FILT PP SUCT OB ISO HV-15768 is open.

D. LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup Pump running, and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 and OB ISO HV-15768 are closed.

Question Data

Answer: B LRW Collection Tank Level trend is level, Suppression Pool Level is constant, Suppression Pool Cleanup pump tripped and SUPP POOL WTR FILT PP SUCT IB ISO HV-15766 is closed.

Explanation/Justification:

A. Suction valve would close stopping transfer to LRW
B. correct answer, Pump trips and IB suction closes on Div I LOCA signal, stopping transfer to LRW.
C. Pump would trip and LRW tank level constant
D. Pump trips, on Div I only IB valve closes.

Sys #	System	Category	KA Statement
295020	**Inadvertent Containment Isolation**	**Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION:**	**Suppression pool water level response**

K/A# **295020.AK3.06** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-159-002**

Question Source: **Modified** **Limerick 1, 1995** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

SSES LOC 19 NRC Exam

27 Unit 2 is operating at 90% power.

- The 2B CRD Pump in service and the in-service Flow Control Valve has failed closed.

Which of the following are the expected alarms for these conditions?

A. AR-204-H05 ROD DRIFT
AR-207-C01 CRD PUMP SUCTION FILTER HIGH DIFF PRESS

B. AR-202-F03 RECIRC PUMP A SEAL CLG WATER LO FLOW
AR-202-F06 RECIRC PUMP B SEAL CLG WATER LO FLOW

C. AR-203-H05 CRD PANEL 1C007 HI TEMP
AR-207-A01, CRD CHARGING WATER HI PRESS

D. AR-207-C02 CRD PUMPS DRIVE WATER FLTR HI DIFF PRESS
AR-207-E02 CRD PUMP B MOTOR OVERLOAD

Question Data

Answer: C AR-203-H05 CRD PANEL 1C007 HI TEMP
AR-207-A01, CRD CHARGING WATER HI PRESS

Explanation/Justification:

A. Flow control valve could cause rod drift if failed open, suction DP due to high flow or strainer clogged.

B. Seal cooling alarm is RBCCW flow.

C. Correct answer, Flow control valve or Drive Pressure Control valve closed will cause alarm and cause decreased cooling water flow.

D. Hi DP caused by high flow or filter clogged, CRD pump will have reduced load and not trip.

Sys #	System	Category	KA Statement
295022	**Loss of CRD Pumps**	**Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:**	**CRD mechanism temperatures**

K/A# **295022.AA2.03** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-207-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

28 Unit 1 is operating at 75% power

- Valve stroke time testing is in progress on the "A" RHR Pump Suppression Pool Suction Valve HV151F004A
- HV151F004A is currently closed
- All other RHR System components are in their normal standby lineup
- A steam break causes Drywell Pressure to reach 2.0 psig and Reactor Pressure to lower to 360 psig.

Assuming no operator actions, which of the following describes the response of the F004A Valve and the "A" RHR Pump?

A. The HV151F004A Valve automatically opens and the "A" RHR Pump will run after the HV151F004A is fully open.

B. The HV151F004A Valve must be opened and the "A" RHR Pump will auto start after HV151F004A is fully open.

C. The HV151F004A Valve automatically opens but the "A" RHR Pump must be started by the operator after HV151F004A is fully open.

D. The HV151F004A Valve must be opened and the "A" RHR Pump must be manually started after HV151F004A is fully open.

Question Data

Answer: B The HV151F004A valve must be opened and the "A" RHR pump will auto start after HV151F004A is fully open.

Explanation/Justification:

A. No auto action associated with the 04 valve.
B. correct answer, valve must be manually opened and pump will auto start since there is Drywell pressure and low reactor pressure
C. No auto action associated with the 04 valve.
D. No auto action associated with the 04 valve, Pump Will auto start with a low pressure or high drywell and low reactor pressure.

Sys #	System	Category	KA Statement
203000	**RHR/LPCI: Injection Mode (Plant Specific)**	**Knowledge of RHR/LPCI: INJECTION MODE design feature(s) and/or interlocks which provide for the following:**	**Adequate pump net positive suction head (interlock suction valve open): Plant-Specific**

K/A# **203000.K4.06** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **Modified** **Hope Creek Unit 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

29 The following conditions exist during refueling operations:

- The Reactor Mode Switch is in REFUEL.
- Irradiated fuel is in the reactor.
- The reactor vessel head has been removed.
- The Fuel Pool gates are installed.
- Reactor water level is 18 inches above the reactor vessel flange.

Which of the following defines the operability status required for shutdown cooling?

A. One shutdown cooling subsystem is required to be operable and in service. The other subsystem is NOT required to be operable.

B. Shutdown cooling is NOT required to be operable with RWCU in service.

C. Two subsystems of shutdown cooling are required to be operable and one subsystem is required to be in service.

D. Two subsystems of shutdown cooling are required to be operable and in service.

Question Data

Answer: C Two subsystems of shutdown cooling are required to be operable and one subsystem is required to be in service.

Explanation/Justification:

A. both shutdown cooling subsystems are required to be operable
B. shutdown cooling is required to be operable
C. correct
D. one subsystem is required to be in operation

Sys #	System	Category	KA Statement
203000	**RHR/LPCI: Injection Mode**	**GENERIC**	**Knowledge of limiting conditions for operations and safety limits**

K/A# 203000.2.2.22 K/A Importance 3.4 Exam Level **RO**

References provided to Candidate **TS 3.9.7 & 3.9.8** Technical References: **SSES Unit 1 Tech Specs 3.9.7**

Question Source: **Modified** **Limerick 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

30 Unit 1 is in cold shutdown with the "A" RHR Pump in the shutdown cooling mode of operation. Reactor water level decreases to the LPCI System initiation setpoint.

What is the expected plant response assuming no operator action five minutes after the LPCI System initiation signal was generated?

A. RHR loop suction and Discharge Valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A" & "C" RHR Pump breakers are cycling open/closed.
The "B" and "D" RHR Pumps are running but not injecting into the reactor vessel.

B. RHR loop suction and Discharge Valves, HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A", "B", "C" and "D" RHR Pumps are injecting into the reactor vessel.

C. RHR loop suction and Discharge Valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO remain open.
The "A" RHR Pump remains in shutdown cooling mode of operation.
The "B", "C" and "D" RHR pumps are injecting into the reactor vessel.

D. RHR loop suction and Discharge Valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO remain open.
The "A" RHR Pump remains in shutdown cooling mode of operation.
The "B", "C" and "D" RHR Pumps are not injecting into the reactor vessel.

Question Data

Answer: A RHR loop suction and discharge valves HV-151-F008 SHUTDOWN CLG SUCT OB ISO, HV-151-F009 SHUTDOWN CLG SUCT IB ISO, HV-151-F015A RHR INJ OB ISO are closed.
The "A" & "C" RHR pump breakers are cycling open/closed.
The "B" and "D" RHR pumps are running but not injecting into the reactor vessel.

Explanation/Justification:

A. correct answer
B. no injection occurs without operator intervention
C. F008, F009, and F015 auto close below +13" reactor water level.
D. F008, F009, and F015 auto close below +13" reactor water level.

Sys #	System	Category	KA Statement
205000	**Shutdown Cooling System (RHR Shutdown Cooling Mode)**	**Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM/MODE including:**	**Pump trips**

K/A# **205000.A3.02** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-149-002, TM-OP-049**

Question Source: **Modified** **Limerick 2, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

31 During a LOCA, HPCI automatically initiated, then tripped. The operator notes the following indications:

- Turbine Stop Valve	Closed
- HPCI Turbine RPM	Zero
- HPCI TURB TRIPPED	Alarm Sealed In
- HPCI TURB TRIP SOLENOID ENERG	Has NOT alarmed

What caused the HPCI turbine to trip?

A. Loss of oil pressure

B. High exhaust pressure

C. High reactor water level

D. Low steam supply pressure

Question Data

Answer: A loss of oil pressure.

Explanation/Justification:

A. Correct answer, Turb Tripped alarm (AR-114-A01) is from valve position, trip solenoid alarm is direct turbine trip, all other distracters are direct Turb trips.

B. Direct Turb trip would cause trip solenoid alarm

C. Direct Turb trip would cause trip solenoid alarm

D. Direct Turb trip would cause trip solenoid alarm

Sys #	System	Category	KA Statement
206000	**HPCI**	**GENERIC**	**Knowledge of annunciator response procedures**

K/A# 206000.2.4.10 K/A Importance 3.0 Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-114-001**

Question Source: **Modified** **Brunswick, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

32 The Core Spray Loop "A" initiation logic channel has experienced a loss of power from 125 VDC Class 1E Bus A (1D614).

Which group of alarms from 1C601 Alarm Panel (AR-109-001) and BIS Display for Core Spray (AR-153-001) would be the expected response to the loss of the 1D614 power supply to the Core Spray Logic?

A. LOOP A OL OR POWER LOSS (AR-153-A01)
LOOP A OUT OF SERVICE (AR-153-B01)

B. LOOP A OL OR POWER LOSS (AR-153-A01)
LOOP A RELAY LGC PWR LOSS (AR-153-A03)

C. LOOP A RELAY LGC PWR LOSS (AR-153-A03)
CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

D. CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)
LOOP A OUT OF SERVICE (AR-153-B01)

Question Data

Answer: C LOOP A RELAY LGC PWR LOSS (AR-153-A03)
CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

Explanation/Justification:

A. Loss of 120VAC control power to any of following valves due to loss of associated power source or breaker racked out.
Thermal overload on any of above valves with CORE SPRAY LOOP A MOV OL BYPS HS-E21-1S12A in TEST.
74 Relay failure on any of above valves.
Loop A out of service caused by the handswitch on the vertical panel not a loss of logic power.

B. LOOP A OL OR POWER LOSS (A01)
Loss of 120VAC control power to any of following valves due to loss of associated power source or breaker racked out.
Thermal overload on any of above valves with CORE SPRAY LOOP A MOV OL BYPS HS-E21-1S12A in TEST.
74 Relay failure on any of above valves.

C. correct answer,

D. Loop A out of service caused by the handswitch on the vertical panel not a loss of logic power.

Sys #	System	Category	KA Statement
209001	**Low Pressure Core Spray System**	**Knowledge of electrical power supplies to the following:**	**Initiation logic**

K/A# **209001.K2.03** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-153-001, AR-109-001**

Question Source: **New** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

33 How does the Automatic Depressurization System (ADS) logic determine that the Core Spray System is available prior to depressurizing the reactor?

A. Both Core Spray Pumps in one loop are operating at normal discharge pressure.

B. At least one Core Spray Pump breaker is closed and the associated loop minimum flow valve is closed.

C. Both Core Spray loop flow rates must be greater than 2000 gpm with the associated minimum flow valve closed.

D. At least one of the two Core Spray Pumps in one loop is operating at normal discharge pressure.

Question Data

Answer: A Both Core Spray Pumps in one loop are operating at normal discharge pressure.

Explanation/Justification:

A. Correct answer
B. Breaker closure is not a permissive.
C. Flow is not a permissive
D. Both pumps to be running with normal discharge pressure.

Sys #	System	Category	KA Statement
209001	**Low Pressure Core Spray System**	**Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following:**	**ADS logic**

K/A# **209001.K3.02** K/A Importance **3.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-83E**

Question Source: **Modified** **FITZPATRICK, 1993** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

34 An ATWS transient occurs and boron must be injected with the Standby Liquid Control System (SLC).

Which of the following describes the effect of a successful initiation of SLC on these indications:

1) Squib Valve Ready (Continuity) lights
2) Alarm on Panel 1C601, SBLC SQUIB VALVES LOSS OF CKT CONTINUITY AR-107-A03
3) Pump discharge pressure

A. 1) Illuminated
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

B. 1) Extinguished
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

C. 1) Illuminated
2) Annunciator not in Alarm.
3) Just above reactor pressure

D. 1) Extinguished
2) Annunciator not in Alarm
3) Just above reactor pressure

Question Data

Answer: B 1) Extinguished
2) Annunciator in Alarm
3) 200 psig greater than reactor pressure

Explanation/Justification:

A. Continuity lights go out.
B. correct answer, Squib valve fires, causing loss of continuity, alarm annunciates indicating loss of continuity, pumps start with discharge pressure slightly greater than reactor pressure.
C. Continuity lights go out. Alarm is annunciated, discharge pressure 200 psig reactor pressure.
D. Alarm is annunciated, discharge pressure 200 psig above reactor pressure.

Sys #	System	Category	KA Statement
211000	**Standby Liquid Control System**	**Ability to manually operate and/or monitor in the control room:**	**System initiation: Plant-Specific**

K/A# **211000.A4.08** K/A Importance **4.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-153-001, TM-OP-053**

Question Source: **Modified** **Duane Arnold 1, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

35 Given the following conditions:

- Unit 2 has experienced a failure-to-scram (ATWS).
- The Standby Liquid Control (SLC) System was initiated and injected for 52 minutes at normal flowrate before both SLC Pumps failed.
- Reactor power is in the Source Range.

How does this failure affect the planned reactor cooldown and depressurization?

A. Reactor Engineering must make the determination if current boron concentration will allow a complete cooldown.

B. Cooldown can be accomplished if completed before Xenon decays out of the core.

C. Reactor boron concentration is sufficient to allow a complete cooldown with a maximum of 8 control rods not fully inserted.

D. Reactor boron concentration is sufficient to allow a complete cooldown under any plant conditions.

Question Data

Answer: D Reactor boron concentration is sufficient to allow a complete cooldown under any plant conditions.

Explanation/Justification:

A. not required.
B. CSBW will account for Xe decay as well.
C. CSBW will handle any number of rods out
D. correct answer, 2 pumps at TS minimum of 41.2 gpm for 52 minutes is 4284 gallons, greater than the CSBW of 4191 gallons.

Sys #	System	Category	KA Statement
211000	**SLC**	**GENERIC**	**Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation**

K/A# **211000.2.1.7** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **EO-000-113**

Question Source: **Modified** **SSES, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

36 During plant shutdown, which of the following Reactor Protection System automatic scrams is bypassed by taking the Mode Switch from RUN to STARTUP?

A. Scram Discharge Volume Level High

B. Main Steam Isolation Valve Closure

C. Turbine Control Valve Fast Closure

D. Turbine Stop Valve Closure

Question Data

Answer: B Main Steam Isolation Valve closure.

Explanation/Justification:

A. not bypassed in startup
B. correct answer
C. bypassed by turbine 1st stage pressure
D. bypassed by turbine 1st stage pressure

Sys #	System	Category	KA Statement
212000	**Reactor Protection System**	**Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following:**	**Main steam system**

K/A# **212000.K1.14** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-58**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

37 With the plant is operating at 80% power, a loss of power to MCC 1B217 occurs.

Which of the below are expected alarms?

A. RPS CHANNEL B1/B2 AUTO SCRAM (AR-104-A01), NEUTRON MON CHAN B SYSTEM TRIP (AR-104-A04), and CORE SPRAY LOOP B OUT OF SERVICE (AR-113-B02)

B. RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP B OUT OF SERVICE (AR-113-B02)

C. RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

D. RPS CHANNEL B1/B2 AUTO SCRAM (AR-104-A01), NEUTRON MON CHAN B SYSTEM TRIP (AR-104-A04)

Question Data

Answer: C RPS CHANNEL A1/A2 AUTO SCRAM (AR-103-A01), NEUTRON MON CHAN A SYSTEM TRIP (AR-103-A04), and CORE SPRAY LOOP A OUT OF SERVICE (AR-109-B02)

Explanation/Justification:

A. Wrong division
B. Core spray valves F004 and 05 powered from 1B217
C. correct answer,
D. Wrong division

Sys #	System	Category	KA Statement
212000	**Reactor Protection System**	**Knowledge of electrical power supplies to the following:**	**RPS motor-generator sets**

K/A# **212000.K2.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-058**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

38 A reactor startup is in progress.

- Power is on Range 2 of the IRMs.
- The "A" IRM increases to 124/125 of scale and the "F" IRM increases to 123/125 of scale.

A Division I half scram occurs.

Which of the following describes the unit RO expected actions?

A. Range up on "A" IRM and reset the Division I side half scram, continue the startup.

B. Range up on "F" IRM and continue the startup.

C. Place the reactor Mode Switch in Shutdown.

D. Range up on "A" and "F" IRM; reset the Division I side half scram, continue the startup.

Question Data

Answer: C Place the reactor Mode Switch in Shutdown.

Explanation/Justification:

A. a full scram should have occurred for these conditions
B. 'F' IRM already above trip setpoint
C. correct
D. a full scram should have occurred for these conditions

Sys #	System	Category	KA Statement
215003	**Intermediate Range Monitor (IRM) System**	**Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including:**	**RPS status**

K/A# **215003.A3.03** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-AD-001**

Question Source: **Modified** **Limerick 1, 1997** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

39 Unit 2 is performing a cold clean startup after a refueling outage.

The 'C' SRM Detector has half of the Argon fill gas pressure as compared to the other operable SRM detectors.

How will the reduced Argon pressure affect the detector response when compared to the other SRM detectors when performing initial SRM/IRM overlap checks?

Will the Reactor Startup be able to continue (assume IRM detectors are on Range 1)?

A. 'C' SRM count rate will be less than the other SRM detectors when IRM indication comes on scale. The 12 hour channel check surveillance will fail but reactor startup will be allowed to continue.

B. 'C' SRM count rate will be less than the other SRM detectors when IRM indication comes on scale. The 24 hour channel check surveillance will fail but reactor startup will be allowed to continue.

C. 'C' SRM count rate will be greater than the other SRM detectors when IRM indication comes on scale. The 12 hour channel check surveillance will fail and reactor startup will not be allowed to continue.

D. 'C' SRM count rate will be the same as the other SRM detectors when IRM indication comes on scale. The 24 hour channel check surveillance will pass and reactor startup will be allowed to continue.

Question Data

Answer: A 'C' SRM count Rate will be less than the other SRM detectors when IRM indication comes on scale. The 12 hour channel check surveillance will fail but reactor startup will be allowed to continue.

Explanation/Justification:

A. *Correct answer, less Argon fill gas reduces the Neutron and Gamma sensitivity of the detector reducing the indicated count rate.* When compared to the other detectors the count rate will be lower and less overlap between the SRM and IRM detectors will be displayed. The Tech Spec Channel check frequency is 12 hours in mode 2 (Startup) and if failed would still allow startup to continue since a minimum of 2 SRMs are required for continued startup.

B. Fission causes recoil within the detector chamber and ionize the Argon gas, decrease in argon gas will reduce sensitivity. Gamma causes direct ionization of argon, decrease in argon gas will reduce sensitivity

C. Fission causes recoil within the detector chamber and ionize the Argon gas, decrease in argon gas will reduce sensitivity.

D. Gamma causes direct ionization of argon, decrease in argon gas will reduce sensitivity

Sys #	System	Category	KA Statement
215004	**Source Range Monitor (SRM) System**	**Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM:**	**Detector operation**

K/A# **215004.K5.01** K/A Importance **2.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TS 3.3.1.2**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

40 - A Unit 2 startup is in progress with power at 20%.
- Recirculation flow is 30%.
- The "A" APRM Flow Unit output remains at 30% as recirculation flow is raised.

As the plant startup continues, what will be the FIRST protective action to occur and the reason for that action?

A. A half scram will occur due to a flow unit INOPERABLE signal.

B. A control rod block will occur due to a flow biased neutron flux upscale.

C. A control rod block will occur due to a flow unit comparator trip.

D. A full scram will occur due to a flow biased neutron flux upscale.

Question Data

Answer: C A control rod block will occur due to a flow unit comparator trip.

Explanation/Justification:

A. flow comparator mismatch or trip is not a half scram signal

B. Flow comparator delta will be first trip signal.

C. Correct, trip set at 10% difference between units. Outputs of the flow comparator units are check for less than 10% difference, in the case given when flow is increased by 10% a trip will occur causing a rod block. For the given conditions the flow biased rod block will occur at 67% power and the scram will occur at 76% power.

D. Flow comparator delta will be first trip signal.

Sys #	System	Category	KA Statement
215005	**Average Power Range Monitor/Local Power Range Monitor System**	**Knowledge of the effect that a loss or malfunction of the following will have on the APRM/LPRM:**	**Flow converter/comparator network: Plant-Specific**

K/A# **215005.K6.07** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-078**

Question Source: **Modified** **Peach Bottom, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

41 Unit 1 at 100% power.

125V DC PANEL 1L610 SYSTEM TROUBLE AR-106-A12 annunciates, indicating a loss of power to Division I 1D614.

How will a loss of Division I 1D614 impact auto initiation and/or steam leak detection isolation of RCIC?

A. Auto initiation will not function if needed. Steam Leak Detection will not function if needed.

B. Auto initiation will function if needed. Steam Leak Detection will function if needed by closing Steam Supply Header Inboard (HV-149-F007) and Warmup Isolation (HV-149-F088) valves.

C. Auto initiation will not function if needed. Steam Leak Detection will function if needed by closing Steam Supply Header Inboard (HV-149-F007) and Warmup Isolation (HV-149-F088) valves.

D. Auto initiation will not function if needed. Steam Leak Detection will function if needed by closing Steam Supply Header Onboard isolation valve (HV-149-F008).

Question Data

Answer: C Auto initiation will not function if needed. Steam leak detection will function if needed by closing steam supply header inboard (HV-149-F007) and warmup isolation (HV-149-F088) valves.

Explanation/Justification:

A. Div II of steam leak detection isolation will function, closing inboard isolation valves.
B. Auto initiation will not function.
C. Correct answer.
D. Div II of steam leak detection isolation will function, not Div I.

Sys #	System	Category	KA Statement
217000	**Reactor Core Isolation Cooling System (RCIC)**	**Knowledge of electrical power supplies to the following:**	**RCIC initiation signals (logic)**

K/A# **217000.K2.02** K/A Importance **2.8** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-050**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

42 The following plant conditions exist:

- A reactor transient is in progress.
- 'B' loop of RHR is in operation.
- 'A' Core Spray pump in operation.
- Drywell pressure 2.5 psig.
- Reactor water level -140 inches.

The following alarm conditions are noted:

RX LO LEVEL SIGNAL A CONFIRMED AR-110-B01 - Alarm window Not lit.
RX LO LEVEL SIGNAL B CONFIRMED AR-110-B03 - Alarm window Lit.
ADS LOGIC C & D TIMER INITIATED AR-110-A03 & AR-110-A04 alarms came in 40 seconds ago.

Which of the following states the response of the ADS to plant conditions with no operator action?

A. 6 ADS valves will open in approximately 60 seconds from Div I & II logic.

B. 3 ADS valves will open in approximately 60 seconds from Div II logic.

C. 3 ADS valves will open in approximately 520 seconds.

D. 6 ADS valves will open in approximately 60 seconds from Div II logic.

Question Data
Answer: D 6 ADS valves will open in approximately 60 seconds from Div II logic.

Explanation/Justification:

A. Div I does not have +13 " confirmation signal nor Div I pp permissive
B. Div II will initiate all ADS valves.
C. Drywell pressure does not require to be bypassed with 7 minute timer.
D. Correct answer, Div I does not have +13 " confirmation signal, Div II will initiate all ADS valves.

Sys #	System	Category	KA Statement
218000	**Automatic Depressurization System**	**Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM:**	**Nuclear boiler instrument system (level indication)**

K/A# **218000.K6.03** K/A Importance **3.8** Exam Level **RO**
References provided to Candidate **None** Technical References: **TM-OP-83**
Question Source: **Modified** **Hope Creek Unit 1, 1995** Level Of Difficulty: (1-5) **3**
Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

43 Which of the statements below is correct regarding the arming and depressing of the four (4) NSSSS pushbuttons on Panel 1C651.

A. "A" and "C" Pushbuttons will cause a full MSIV isolation.

B. "A" and "B" Pushbuttons will cause a full inboard isolation including all 8 MSIVs.

C. "B" and "C" Pushbuttons will cause an inboard MSIV isolation.

D. "A" and "D" Pushbuttons will cause an outboard MSIV isolation.

Question Data

Answer: **B** "A" and "B" pushbuttons will cause a full inboard isolation including all 8 MSIVs.

Explanation/Justification:

A. cause no isolation
B. correct answer
C. 'B' & 'C' button causes an inboard isolation, inboard and outboard MSIV isolation
D. 'A' & 'D' causes a full inboard and outboard isolation

Sys #	System	Category	KA Statement
223002	**Primary Containment Isolation System/Nuclear Steam Supply Shut-Off**	**Knowledge of PCIS/NSSSS design feature(s) and/or interlocks which provide for the following:**	**Manual initiation capability: Plant-Specific**

K/A# **223002.K4.03** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-059**

Question Source: **Modified** **SUSQUEHANNA, 1989** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

44 With the reactor at 100% power, which of the following would be an indication of an open Safety Relief Valve (SRV)?

A. Total indicated steam flow increase.

B. Indicated Feed flow less than Steam flow.

C. SRV Tailpipe temperature stable at 500 deg F.

D. Reactor thermal power increase.

Question Data

Answer: D Reactor thermal power increase.

Explanation/Justification:

A. Indicated total steam flow will decrease.
B. Feed flow will indicate greater than steam flow. The candidate may think if actual steam flow increases, then feed flow will increase.
C. Tail pipe temperature will peak at approximately 320 degrees.
D. correct answer, power will increase slightly due to reduced feed heating due to loss of extraction steam heating.

Sys #	System	Category	KA Statement
239002	**Relief/Safety Valves**	**Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including:**	**Reactor power**

K/A# **239002.A1.06** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-183-001**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

45 Unit 1 is at 84% power with Feedwater Control in 'Average Level Control' and Three-Element Control.

What would be the response if the Unit 1 "A" Narrow Range Level instrument were equalized (Assume no operator action)?

A. All three feed pumps and the main turbine trip.

B. RPV level will lower to approximately 23 inches, and remain steady at the new lower level.

C. Division I scram on low level, High level for Division II.

D. RPV level will rise to approximately 47 inches, and remain steady at the new higher level.

Question Data

Answer: B RPV level will lower to approximately 23 inches and remain steady at the new lower level.

Explanation/Justification:

A. signal only affected one instrument, need two out of three to trip turbines

B. correct answer, zero delta p is a failure upscale.

C. RPV level will not lower to scram setpoint because of average level circuit

D. the zero DP signal will simulate a high level signal and cause FW control to lower level not control higher.

Sys #	System	Category	KA Statement
259002	**Reactor Water Level Control System**	**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 reactor water level input**

K/A# 259002.A2.03 K/A Importance 3.3 Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-045**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

46 The following conditions exist:
- Unit 2 is operating at 100% power.
- The Standby Gas Treatment System (SGTS) is in the standby lineup.
- A valid Unit 2 high drywell pressure initiation signal occurs.

Which of the following are the locations from which the SGTS will automatically take suction.

A. U-2 Drywell, Reactor Building Zone 2 and Reactor Building Zone 3

B. Reactor Building Zone 2 and Reactor Building Zone 3; Unit 2 HPCI Barometric Condenser

C. U-2 Drywell and U-2 HPCI Barometric Condenser

D. Only the U-2 Reactor Building Zone 2 and Reactor Building Zone 3

Question Data

Answer: B Reactor Building Zone 2 and Reactor Building Zone 3, Unit 2 HPCI Barometric Condenser.

Explanation/Justification:

A. U-2 Drywell does not align automatically
B. correct answer
C. U-2 Drywell does not align automatically
D. HPCI Barometric Condenser exhausts to the SBGTS suction.

Sys #	System	Category	KA Statement
261000	**Standby Gas Treatment System**	**Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following:**	**Primary containment pressure**

K/A# **261000.K1.11** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-073**

Question Source: **Modified** **Browns Ferry 1, 2, & 3, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

47 Which of the following ESS 4 kV bus feeder breaker trips will NOT auto transfer the bus to its alternate source?

A. Incoming feeder overcurrent

B. ESS Transformer pressure relay

C. Undervoltage

D. ESS Transformer differential current

Question Data

Answer: A Incoming feeder overcurrent

Explanation/Justification:

A. correct answer,
B. transformer lockout signal only
C. causes transfer to alternate source
D. transformer lockout signal only

Sys #	System	Category	KA Statement
262001	**A.C. Electrical Distribution**	**Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following:**	**Off-site power system**

K/A# **262001.K3.05** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-004**

Question Source: **Modified** **Peach Bottom, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

48 The 1E Instrument AC UPS Distribution 1D130 supplies power to 1Y128 and 0Y301.

Immediately after the preferred source 1B246 is lost, which of the following will be the correct electrical configuration for 1D130 to supply power to 1Y128 and 0Y301?

A. 1B226, Alternate source stepped down to 120VAC

B. Battery, 1D133 inverted to 120VAC

C. Battery, 1D143 inverted to 120VAC

D. 1B226, Maintenance Backup, rectified to 120VDC, inverted to 120VAC

Question Data

Answer: B Battery, 1D133 inverted to 120VAC

Explanation/Justification:

A. Battery is first alternate power supply, until dead then 1B226 is the power supply.

B. Correct answer, battery, UPS being supplied from the dedicated battery can carry the distribution panels for approximately 20 minutes. When the external battery's output decreases to less than 210 VDC, the Static Transfer switch operates to supply alternate power to the distribution panels, and alternate supply is via 480-208/120 V step down transformers

C. incorrect, there is no 1D143 battery

D. Maintenance backup is used for maintenance and is a manual transfer.

Sys #	System	Category	KA Statement
262002	**Uninterruptable Power Supply (A.C./D.C.)**	**Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following:**	**Transfer from preferred power to alternate power supplies**

K/A# **262002.K4.01** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-017**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

49 With no operator action, which one of the following functions are lost as a result of a loss of 125V DC Bus 1D624?

A. "D" Diesel Generator START CONTROL SEQUENCE.

B. "A" or "C" Core Spray Pumps auto start.

C. Division 1 SPOTMOS monitoring.

D. "D" ESW Pump auto start.

Question Data

Answer: D "D" ESW pump auto start

Explanation/Justification:

- **A.** Div II D D/G from 1D644
- **B.** Div I Core Spray logic from 1D614
- **C.** Div 1 SPOTMOS from 1D614
- **D.** correct answer, Attachment A item 9 and Attachment B BKR 36

Sys #	System	Category	KA Statement
263000	**D.C. Electrical Distribution**	**Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following:**	**Components using D.C. control power (i.e. breakers)**

K/A# **263000.K3.02** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **ON-102-620** Technical References: **ON-102-620**

Question Source: **Bank** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

50 During normal operation, the AC power supply to the 1D663 Battery Charger is lost.

Which of the following describe the automatic actions for the loss of the AC power to this battery charger?

A. The alternate battery charger immediately feeds the bus,
250V DC PANEL 1L650 SYSTEM TROUBLE AR-106-A11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

B. The battery charger is immediately supplied by alternate AC power,
250V DC PANEL 1L650 SYSTEM TROUBLE AR-106-A11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

C. The battery backfeeds the inverter and forward feeds the bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-106-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

D. The battery immediately feeds the 250 VDC bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-106-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

Question Data

Answer: D The battery immediately feeds the 250 VDC bus,
250V DC PANEL 1L660 SYSTEM TROUBLE AR-106-B11 alarms on 1C651,
BATTERY CHARGER TROUBLE alarms on local alarm panel.

Explanation/Justification:

A. No alternate battery charger on Div II, only on Div I.
B. No alternate AC power supply for battery charger.
C. Battery feeds DC distribution bus directly.
D. correct answer, Control room alarm received from local alarm panel, local alarm panel alarm due to AC supply to battery charger lost.

Sys #	System	Category	KA Statement
263000	**D.C. Electrical Distribution**	**Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including:**	**Meters, dials, recorders, alarms, and indicating lights**

K/A# **263000.A3.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-106-001**

Question Source: **Modified** **Browns Ferry 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

51 The A Diesel Generator (DG) is being run for the monthly operability check.

The PCOP has just closed the diesel output breaker to parallel the DG to the bus, when the following is noted:

- DG Frequency is 60 Hz.
- DG Kilowatts is 5 kW.
- DG Kilovars is -4500 Kvar.

Attempting to lower Kvars on the DG, the operator takes the SPEED ADJUST switch to lower. SELECT the DG response to this action:

A. The DG will trip on loss of field voltage.

B. Frequency will decrease rapidly.

C. The DG will trip on reverse power.

D. The DG will slip a pole.

Question Data

Answer: C The DG will trip on reverse power.

Explanation/Justification:

A. trip on reverse power

B. Frequency can't change with D/G tied to grid

C. correct answer, D/G not loaded any decrease in fuel will reduce load further.

D. trip on reverse power not slip a pole

Sys #	System	Category	KA Statement
264000	**Emergency Generators (Diesel/Jet)**	**Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including:**	**Maintaining minimum load on emergency generator (to prevent reverse power)**

K/A# **264000.A1.09** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **SO-024-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

52 Maintenance is requesting that both Containment Instrument Gas (CIG) compressors be removed from service due to a common mode failure potential. The plan requires that Instrument Air be lined up to the CIG System for approximately 12 hours while the compressors are repaired.

During CIG compressor maintenance, Instrument Air is lost due to a system breach.

How would the CIG System and the CIG supplied components respond to this event?

A. Pressure for the 90# CIG Header from the storage bottles, no valves repositioning.

B. No pressure for the 90# CIG Header, Closure of Inboard MSIVs, No pressure for the 150# CIG header, Safety Relief Valves work only in Safety function.

C. No pressure for the 90# CIG Header, Closure of Inboard MSIVs, loss of Drywell Cooling.

D. Pressure for the 90# and 150# CIG Header from the backup Service Air tie, no valves repositioning.

Question Data

Answer: C No pressure for the 90# CIG header, Closure of Inboard MSIVs, loss of Drywell Cooling.

Explanation/Justification:

A. 150# storage bottles can not feed the 90# header.

B. 150# header would be supplied with the storage bottles for longer than 12 hours.

C. Correct answer, if Instrument air is lost, there is no pressure for the 90# header. The 90# header can not be supplied from any of the 150# header due to check valve arrangement. The outlet of the CIG compressors split off to the 90# header and the 150 # header. Instrument air ties into the 90# header in such a manner that it can not back feed to the 150# header. On a loss of Instrument air there is no way to keep the containment isolation valves open MSIV, Chilled water.

D. There is not CIG/Service air cross tie, and supply to 90# header can not supply 150# header.

Sys #	System	Category	KA Statement
300000	**Instrument Air System (IAS)**	**Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following:**	**Containment air system**

K/A# **300000.K3.01** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-025**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

53 Unit 1 is at 100% power when a pipe break occurs on the Unit 1 TBCCW inlet to the in service heat exchanger.

What alarms and plant indications would be expected as a result of the pipe break?

A. TBCCW HEAD TANK HI-LO LEVEL AR-123-G06
TBCCW HX AREA FLOODED AR-123-H01
TBCCW Heat Exchanger Discharge Pressure Indication (PI-14409) downscale
Off gas precoolers Hi temperature

B. TBCCW PUMPS DISCHARGE HEADER LO PRESS AR-123-G03
TBCCW HEAT EXCHANGER HEADER LO PRESS AR-123-G04
Standby TBCCW Pump running
Instrument Air Compressors tripped

C. TBCCW PUMPS DISCHARGE HEADER LO PRESS AR-123-G03
TBCCW HEAD TANK HI-LO LEVEL AR-123-G06
TBCCW HX AREA FLOODED (H01)
PASS Sample cooler Hi temperature

D. TBCCW HX AREA FLOODED AR-123-H01
TBCCW Heat Exchanger Discharge Pressure Indication (PI-14409) downscale
Standby TBCCW Pump running
RFPT Lube Oil Cooler Hi temperature

Question Data

Answer: B TBCCW PUMPS DISCHARGE HEADER LO PRESS G03
TBCCW HEAT EXCHANGER HEADER LO PRESS AR-123-G04
Standby TBCCW pump running
Instrument Air Compressors tripped

Explanation/Justification:

A. Offgas precooler RBCCW
B. Correct answer
C. Pass cooler RBCCW
D. RFPT Lube Oil Cooler Service water

Sys #	System	Category	KA Statement
400000	**Component Cooling Water System (CCWS)**	**Ability to manually operate and/or monitor in the control room:**	**CCW indications and control**

K/A# **400000.A4.01** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-115-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

54 The reactor is at 100% power. CRD PUMP 1P132B is out-of -ervice. Annunciator "CRD PUMP 'A' TRIP " alarms. What additional alarms will annunciate as a result of the pump trip?

How is the movement of the control rods from the control room affected?

A. ACCUMULATOR TROUBLE (AR-103-H06) alarms and Rods can be scrammed and withdrawn, but not individually inserted.

B. CRD MECHANISM TEMP (AR-103-H05) and RECIRC PUMP A SEAL CLG WATER LO FLOW (AR-102-F03) alarms and Rods can be scrammed and inserted, but not individually withdrawn.

C. RWCU PUMP TROUBLE (AR-101-D01) alarms and Rods can only be scrammed.

D. CRD MECHANISM TEMP (AR-103-H05) and RECIRC PUMP A SEAL CLG WATER LO FLOW (AR-102-F03) alarms and Rods can only be scrammed.

Question Data

Answer: C RWCU PUMP TROUBLE (AR-101-D01) alarms and Rods can only be scrammed.

Explanation/Justification:

A. No insert or withdraw available

B. No insert or withdraw available

C. correct answer, Scram Accumulator pressure assures that all functions work, although rod motion is slow, and CRD mechanism cooling is lost.

D. Recirc pump a seal cooling water low flow is not associated with CRD system.

Sys #	System	Category	KA Statement
201001	**Control Rod Drive Hydraulic System**	**Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System:**	**A.C. power**

K/A# **201001.K6.05** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-155-007**

Question Source: **Modified** **Duane Arnold, 1996** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

55 Select the method by which reactivity insertion rate is regulated for control rod withdrawals and insertions.

Reactivity insertion rate is controlled by:

A. throttling the water flow entering and leaving the over piston area of the drive mechanism.

B. automatically varying the position of the Control Rod Drive Hydraulic Pressure Control Valve.

C. automatically varying the operating position of the Control Rod Drive Hydraulic Flow Control Valve.

D. throttling the water flow entering and leaving the under piston area of the drive mechanism.

Question Data

Answer: D throttling the water flow entering and leaving the under-piston area of the drive mechanism.

Explanation/Justification:

A. over-piston area flow is leaving is not throttled.
B. pressure control valve is not automatically positioned.
C. flow control valve controls total system flow not regulated flow to each mechanism
D. correct answer

Sys #	System	Category	KA Statement
201003	**Control Rod and Drive Mechanism**	**Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including:**	**CRD drive water flow**

K/A# **201003.A1.03** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-055**

Question Source: **Modified** **River Bend 1, 1997** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

56 What is the design bases for allowing the Rod Sequence Control System to be automatically bypassed above 20% power?

A. No combination of operator errors could result in fuel damage due to a Control Rod Drop Accident.

B. The power excursion caused by a Control Rod Drop Accident would be terminated by an APRM hi-flux SCRAM.

C. The RBM prevents any control rod from attaining the rod worth necessary to damage fuel on a Control Rod Drop Accident.

D. Rodworth Minimizer will continue to enforce the rod control sequence.

Question Data

Answer: A No combination of Operator errors could result in fuel damage due to a Control Rod Drop Accident.

Explanation/Justification:

A. correct answer, voids in large enough quantity to minimize differential rod worth.
B. possibly but not the basis
C. RBM minimizes local power increase by limiting amount of power change, does not affect rod worth
D. RWM also bypassed

Sys #	System	Category	KA Statement
201004	**Rod Sequence Control System (Plant Specific)**	**Knowledge of the operational implications of the following concepts as they apply to ROD SEQUENCE CONTROL SYSTEM:**	**Prevention of clad damage if a control rod drop accident (CRDA) occurs: BWR-4, 5**

K/A# **201004.K5.01** K/A Importance **3.6** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-056**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

57 With the Rod Worth Minimizer keylock in NORMAL, a loss of rod position signal from the selected rod to the Rod Worth Minimizer will cause:

A. only a withdraw block if power is less than the Low Power Setpoint.

B. only a "SYSTEM ERROR" display.

C. a withdraw and insert rod block if power is less than the Low Power Setpoint.

D. a withdraw and insert rod block at any power.

Question Data

Answer: C a withdraw and insert rod block if power is less than the Low Power Set Point.

Explanation/Justification:

A. also insert block
B. withdraw and insert block
C. correct answer
D. blocks are bypassed above 20% power

Sys #	System	Category	KA Statement
201006	**Rod Worth Minimizer System (RWM) (Plant Specific)**	**Ability to predict and/or monitor changes in parameters associated with operating the ROD WORTH MINIMIZER SYSTEM (RWM) controls including:**	**Rod position: P-Spec(Not-BWR6)**

K/A# **201006.A1.01** K/A Importance **3.2** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-031D**

Question Source: **Modified** **Susquehanna 1 & 2, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

58 A Design Basis Accident has occurred on Unit 1 with a lockout on the 1A201 ESS Bus.

Which of the following describes the impact, and why?

A. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will not operate, power is lost from 1B219.

B. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with power from 1B219.

C. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with alternate power from 1B229.

D. RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will not operate, power is lost from 1B229.

Question Data

Answer: B RHR Outboard Injection Vlv HV-151-F015A and Reactor Recirc Pump "A" Discharge Vlv HV-143-F031A will operate with power from 1B219.

Explanation/Justification:

A. ATS will supply power from 1B230.

B. Correct answer, ESS Load Center 1B210 is normal power supply via Swing Bus MG Set for 1B219, loss of power from ESS bus 1A201 to 1B210 will cause ATS to seek power from 1B230.
The 1B219 alternate source is ESS Load Center 1B230

C. 1B229 is power supply for Div 2 valves.

D. 1B229 is power supply for Div 2 valves.

Sys #	System	Category	KA Statement
202001	**Recirculation System**	**Knowledge of electrical power supplies to the following:**	**Recirculation system valves**

K/A# **202001.K2.03** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-004**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

59 While making a tour of the Unit 1 Lower Relay Room, you notice an alarm light on a RBM Channel on top of Panel 1C608. The light is labeled, "REF DNSCL".

What RBM function is associated with this alarm?

A. Automatically bypasses RBM

B. Initiates Rod Block

C. Bypasses Rod Insert Blocks

D. Indication only for the reference APRM at 30% power

Question Data

Answer: A Automatically bypasses RBM

Explanation/Justification:

A. correct
B. Not a RBM rod block
C. RBM does not block rod insertion
D. provides control function as well as indication.

Sys #	System	Category	KA Statement
215002	**Rod Block Monitor System**	**Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including:**	**Alarm and indicating lights: BWR-3, 4, 5**

K/A# **215002.A3.03** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-078**

Question Source: **Modified** **Dresden 2 & 3, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.41**

60 Unit 1 is operating at 100% power.

- Wide Range level indicates +50 inches.
- RX WATER HI LEVEL AR-101-A17 alarm on 1C651.
- Green light above the LEVEL LOGIC RESET A HS-C32-1S04A Switch is lit on 1C651.
- Green light above the LEVEL LOGIC RESET B HS-C32-1S04B Switch is lit on 1C651.
- Green light above the LEVEL LOGIC RESET C HS-C32-1S04C Switch is NOT lit on 1C651.

Which of the following is the expected response for the above indications

A. No Reactor Feed Pumps tripped, no HPCI or RCIC trip alarms.

B. 'A', 'B' & 'C' Reactor Feed Pumps tripped, no HPCI or RCIC trip alarms.

C. 'A' & 'B' Reactor Feed Pumps tripped, 'C' Reactor Feed Pump feeding, trip alarms annunciated for HPCI or RCIC

D. No Reactor Feed Pumps tripped, trip alarms annunciated for HPCI or RCIC

Question Data

Answer: B 'A', 'B' & 'C' Reactor Feedpumps tripped, no HPCI or RCIC trip alarms.

Explanation/Justification:

A. Trip logic for RFPs is actuated.
B. correct answer, two out of three logic for trip of feedpumps, HPCI-RCIC hi level trip come from different level switches thus possible not tripped.
C. No other alarms as mentioned in stem, thus no HPCI/RCIC alarms
D. RFPs tripped, no alarms annunciated for HPCI/RCIC

Sys #	System	Category	KA Statement
216000	**Nuclear Boiler Instrumentation**	**Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following:**	**Protection against filling the main steam lines from the feed system**

K/A# **216000.K4.09** K/A Importance **3.3** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-080**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

61 Suppression Pool cooling is in service on Unit 2 with the 2A RHR Pump in service when an electrical fault causes Bus 2A201 to de-energize.

All other buses remain energized.

Which of the following describes how Suppression Pool cooling will be affected by this bus loss?

A. The 2D RHR Pump will be de-energized; Loops A & B are available for Suppression Pool Cooling.

B. The 2A and 2C RHR Pumps will be de-energized; Loop B is available for Suppression Pool Cooling.

C. The 2A RHR Pump will be de-energized; Loop B is available for Suppression Pool Cooling.

D. The 2B and 2D RHR Pumps will be de-energized; Loop A is available for Suppression Pool Cooling.

Question Data

Answer: C The 2A RHR Pump will be deenergized; Loop B is available for Suppression Pool Cooling.

Explanation/Justification:

A. only '2A' pump not available.
B. only '2A' pump not available.
C. correct answer, '2A' pump powered from 2A ESS bus.
D. only '2A' pump not available.

Sys #	System	Category	KA Statement
219000	**RHR/LPCI: Torus/Suppression Pool Cooling Mode**	**Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE controls including:**	**System flow**

K/A# **219000.A1.02** K/A Importance **3.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-049, ON-204-210**

Question Source: **Modified** **Quad Cities 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

62 Following an Auxiliary Bus Load Shed with all of the emergency buses energized, which of the following components will require operator actions to provide it with a source of cooling water?

A. Diesel Generators

B. Condensate Pumps

C. HPCI Room Coolers

D. Diesel Driven Fire Pump

Question Data

Answer: B Condensate pumps

Explanation/Justification:

A. No additional operator actions required, ESW cooled.

B. Correct answer, TBCCW available, due to Aux Bus Load Shed no service water available to cool component cooling water. Restoration of Aux Load shed procedure (ON-104-001) instructs ESW lined up to substitute for Service water.

C. No additional operator actions required, ESW cooled.

D. No additional operator actions required, self-cooled with radiator

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Sys #	System	Category	KA Statement
256000	**Reactor Condensate System**	**Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Loss of equipment component cooling water systems**

K/A# **256000.A2.12** K/A Importance **3.1** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-104-001**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

SSES LOC 19 NRC Exam

63 Unit 1 & 2 are operating at 100% power when the following annunciator is received in the Control Room:

REFUEL FLOOR WALL EXHAUST HI-HI RADIATION (AR-101-A05)

Which of the following is the expected ventilation response?

A. Isolation of Reactor Building Zone III Ventilation.
Automatic start of both SGTS fans.
RB Zone III Filtered Exh Fans 1V217A(B) and 2V217A(B) start.

B. Isolation of Reactor Building Zone III Ventilation.
Automatic start of both SGTS fans.
Automatic start of Reactor Building Zone III Recirculation System.

C. SGTS Train A or B start.
Emergency Outside Air Supply Fan A(B) starts.
RB Recirc System to SBGT HD07543A(B) closes.

D. SGTS Train A or B start.
RB Zone III Iso Dampers HD27502A(B), HD27514A(B) HD27564A(B), HD17502A(B), HD17514A(B) and HD17564A(B) closes.
RB Zone III Filtered Exh Fans 1V217A(B) and 2V217A(B) start.

Question Data

Answer: B Isolation of Reactor Building Zone III Ventilation.
Automatic start of both SGTS fans.
Automatic start of Reactor Building Zone III Recirculation System.

Explanation/Justification:

A. RB Zone III Filtered Exh Fans trip not start.
B. Correct answer
C. RB Recirc System to SBGT HD07543A(B) open instead of closing.
D. RB Zone III Filtered Exh Fans trip not start.

Sys #	System	Category	KA Statement
272000	**Radiation Monitoring System**	**Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including:**	**Ventilation system isolation indications**

K/A# **272000.A3.06** K/A Importance **3.4** Exam Level **RO**

References provided to Candidate **None** Technical References: **TM-OP-079**

Question Source: **Modified** **Big Rock Point 1, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

64 A loss of Zones I and III ventilation has occurred with Unit 1 at 100% power. The loss of Reactor Building Ventilation Off-Normal Procedure has been entered, room cooling initiated, and Maintenance support requested.

Which area is expected to have the most rapid temperature increase?

A. ESS Switchgear Rooms

B. HPCI Room

C. Main Steam Pipe Tunnel

D. RWCU Pump Room

Question Data

Answer: D RWCU Pump Room

Explanation/Justification:

A. ESS Switchgear rooms has its own cooling system which remains in service with a loss of Zone I HVAC.
B. HPCI Room has it's own cooling system which remains in service with a loss of Zone I HVAC.
C. Main Steam Pipe Tunnel has it's own cooling system which remains in service with a loss of Zone I HVAC.
D. Correct answer, The RWCU area does not have a room cooler and is cooled by air flow provided by the Rx Bldg HVAC. A loss of HVAC will cause the room temperature to rise rapidly.

Sys #	System	Category	KA Statement
288000	**Plant Ventilation Systems**	**Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on following:**	**Reactor building temperature: Plant-Specific**

K/A# **288000.K3.02** K/A Importance **2.9** Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-134-002**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

65 Unit 1 had a loss of Drywell Cooling while operating at 100% power due to a spurious high Drywell pressure signal.

The high Drywell pressure signal is cleared.

What procedural steps need to be completed to reset Drywell Cooling isolation logic?

A. Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Go to open for A & B Drywell Cooler Inboard and Outboard Isolation Valves

B. Go to close for A & B Drywell Cooler Inboard and Outboard Isolation Valves
Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681

C. Reset MN STM LINE DIV I & II logic on 1C601
Go to close for A & B Drywell Cooler Inboard and Outboard Isolation Valves
Reset Div I & II RBCW ISO VALVE POS logic on 1C681

D. Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Verify A & B Drywell Cooler Inboard and Outboard Isolation Valves open

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Question Data

Answer: D Reset Div I & II DRWL CLG logic on 1C601
Reset Div I & II RBCW ISO VALVE POS logic on 1C681
Verify A & B Drywell Cooler inboard and outboard isolation valves open

Explanation/Justification:

A. No need to open valves
B. No need to go to close.
C. No need to go to close
D. Correct answer

Sys #	System	Category	KA Statement
290001	**Secondary Containment**	**Ability to manually operate and/or monitor in the control room:**	**System reset: Plant-Specific**

K/A# 290001.A4.11 K/A Importance 3.4 Exam Level **RO**

References provided to Candidate **None** Technical References: **ON-159-002**

Question Source: **Modified** **Hope Creek Unit 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

66 The US, PCO, and I&C have discussed the performance of a Surveillance Procedure on the ADS System. The annunciators expected to be received during the surveillance have been identified and reviewed.

Which of the following describes the required actions when one of the expected annunciators is received?

The operator shall acknowledge the alarm and:

A. is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.

B. is NOT required to report the alarm to the US. The associated alarm response procedure shall be checked.

C. the annunciator shall be reported to the US. The operator does NOT have to refer to the associated alarm response procedure.

D. the annunciator shall be reported to the US. The associated alarm response procedure shall be checked.

Question Data

Answer: A is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.

Explanation/Justification:

A. correct answer, The operator shall acknowledge the alarm and is NOT required to report the alarm to the US. The operator does NOT have to refer to the associated alarm response procedure.
B. The Alarm Procedure does not have to be referred to under these conditions
C. No report is required to the US
D. No report is required to the US

Sys #	System	Category	KA Statement
		Conduct of Operations	**Knowledge of operator responsibilities during all modes of plant operation.**

K/A# **2.1.2** K/A Importance **3.0** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-AD-004 (Sect 11.2.c)**

Question Source: **Modified** **Cooper 1, 1999** Level Of Difficulty: (1-5) **2**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

67 A Unit 1 Startup is in progress per the GO-100-002, Plant Startup, Heatup and Power Operation. Prior to commencing the heatup, the Unit Supervisor directs Reactor Vessel water level be maintained in a band of 30 to 40 inches with RWCU in accordance with OP-161-001.

After initial setup of RWCU no operator actions are taken.

How will the RWCU System respond as the Reactor is heated up?

A. Regenerative Heat Exchanger inlet temperature will increase.
Non-Regenerative Heat Exchanger inlet temperature will remain constant.
The RBCCW to RWCU will isolate on High Temperature.
The RWCU System will isolate on high temperature.

B. Regenerative Heat Exchanger outlet temperature will increase.
Non-Regenerative Heat Exchanger outlet temperature will remain constant.
Flow through the RWCU Blowdown Flow Control Valve (HV-144-F033) will be controlled constant.
Pressure downstream of the RWCU Blowdown Flow Control Valve (HV-144-F033) will be constant.

C. Regenerative Heat Exchanger inlet temperature will increase.
Non-Regenerative Heat Exchanger inlet temperature will increase.
The RWCU Blowdown Flow Control Valve (HV-144-F033) will auto close on high upstream pressure.
The Blowdown Valve to the Condenser Hotwell (HV-144-F034) will auto close on high upstream pressure.

D. Regenerative Heat Exchanger outlet temperature will increase.
Non-Regenerative Heat Exchanger outlet temperature will increase.
Flow through the RWCU Blowdown Flow Control Valve (HV-144-F033) will increase.
Pressure downstream of the RWCU Blowdown Flow Control Valve (HV-144-F033) will increase.

Question Data

Answer: D Regenerative Heat Exchanger outlet temperature will increase
Non- Regenerative Heat Exchanger outlet temperature will increase
Flow through the RWCU blowdown flow control valve (HV-144-F033) will increase
Pressure downstream of the RWCU blowdown flow control valve (HV-144-F033) will increase

Explanation/Justification:

A. RBCCW does not isolate on high temperature, a loss of drywell cooling signal will causes RBCCW to isolate.

B. The flow control valve does not control with a flow signal but manually by position. The downstream pressure would remain relatively constant if valve position remained constant.

C. The blowdown valve closes on high pressure down stream and low pressure upstream.

D. correct answer, RWCU is aligned during startup for level control by draining to Rad Waste or the Main Condenser. The drain path takes water from the outlet of the RWCU filters which is cool water prior to the Regen Hx, The blowdown water is cooling water for the Regen Hx that is short-circuiting the Regen Hx causing outlet temperature to increase. Hotter water into the Non-regen means higher outlet temperature. As the plant heat-up continues, pressure increases causing a larger differential pressure across the blowdown valve causing an increase of flow.

Sys #	System	Category	KA Statement

Ability to perform specific system and integrated plant procedures during different modes of plant operation.

K/A# 2.1.23 K/A Importance 3.9 Exam Level RO

References provided to Candidate **None** Technical References: **OP-161-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

68 During a startup on Unit 2 reactor, the Plant Control Operator withdraws Control Rod 26-27 from Notch 32 to Notch 34, reactor period changes from 200 seconds to a stable 50 second period.

Which of the following identifies the required action to be taken?

A. Re-insert control rods as necessary to achieve sub-criticality.

B. Shut down the reactor until a thorough assessment has been performed.

C. Re-insert ControlRrod 26-27 to obtain a stable period indication of greater than 100 seconds.

D. Do not move any additional rods until a Core Monitor is run.

Question Data

Answer: C Re-insert control rod 26-27 to obtain a stable period indication of greater than 100 seconds.

Explanation/Justification:

A. obtain a stable period indication of greater than 100 seconds.
B. obtain a stable period indication of greater than 100 seconds.
C. correct answer
D. obtain a stable period indication of greater than 100 seconds.

Sys #	System	Category	KA Statement
		Conduct of Operations	**Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and *instrument interpretation*.**

K/A# **2.1.7** K/A Importance **3.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **GO-200-002, 6.23**

Question Source: **Modified** **Browns Ferry 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

69 Given the following conditions:

- Reactor power is 40%
- ALL Turbine Bypass Valves fail OPEN
- The MSIVs FAIL to automatically close
- The MSIVs are closed manually

Prior to MSIV closure, which of the following combinations of reactor power and reactor pressure would indicate a safety limit violation had occurred?

A. Reactor power is 30% and RPV pressure is 810 psig.

B. Reactor power is 30% and RPV pressure is 775 psig.

C. Reactor power is 20% and RPV pressure is 795 psig.

D. Reactor power is 10% and RPV pressure is 810 psig.

Question Data

Answer: B Reactor power is 30% and RPV pressure is 775 psig.

Explanation/Justification:

A. Pressure and flow within Safety Limit

B. correct answer, <785 psig, <10E6 lbm core flow, thermal power must be < 25%.

C. Power within Safety Limit of 25%

D. Pressure, power and flow within Safety Limit.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of limiting conditions for operations and safety limits.**

K/A# 2.2.22 K/A Importance 3.4 Exam Level **RO**

References provided to Candidate **None** Technical References: **T.S. 2.1**

Question Source: **Modified** **Dresden 2, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

SSES LOC 19 NRC Exam

70 A Procedure Change for immediate use is required for RCIC Quarterly Flow Verification, SO-150-002. The HV-149-F022 TEST LINE ISO TO CST needs to be set to a position other than 40% OPEN as required by the procedure because of a pump impeller modification.

Which of the following steps must be completed prior to use of the procedure change?

A. Obtain PORC Committee review, Log PCAF in the Control Room Procedure PCAF Log, Stamp PCAF placed in Controlled Manuals as CONTROLLED, deliver the original PCAF to Training.

B. Obtain responsible Functional Unit Manager approval, Stamp PCAF placed in Controlled Manuals as CONTROLLED, QC approve insertion into appropriate manuals.

C. Log PCAF in the Control Rroom Procedure PCAF Log, Stamp PCAF placed in Controlled Manuals as CONTROLLED, deliver the original PCAF to DCS, Obtain PORC committee review.

D. Obtain responsible Functional Unit Manager approval, Stamp PCAF placed in Controlled Manuals as CONTROLLED, deliver the original PCAF to DCS.

Question Data

Answer: C Log PCAF in the control room Procedure PCAF Log, Stamp PCAF placed in controlled manuals as CONTROLLED, Deliver the original PCAF to DCS, Obtain PORC committee review.

Explanation/Justification:

A. PCAF is delivered to DCS.
B. QC does not approve PCAFs
C. Correct answer.
D. PCAF requires PORC approval.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of the process for controlling temporary changes.**

K/A# **2.2.11** K/A Importance **2.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **NDAP-QA-0002 (8.3)**

Question Source: **Modified** **Quad Cities 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

71 Unit 1 is in a refuel outage.

Operations, Reactor Engineering (RE) and In-Service Inspection (ISI) Engineers are moving Control Rods and moving an underwater camera to verify Fuel Channel to Control Rod clearances.

Division II SRM detectors are withdrawn for I&C work.

Division I 24 VDC power supply for the SRMs is lost.

How will the status of SRM detectors impact this evolution?

A. All Control Rod movement must stop, ISI camera movement required to stop.

B. All Control Rod movement must stop, ISI camera movement may continue.

C. Control Rod withdrawal with fuel in associated control cell must stop, ISI camera movement must also stop.

D. Control Rod withdrawal with fuel in the associated control cell MUST stop, ISI camera movement may continue.

Question Data

Answer: D Control Rod withdrawal with fuel in the associated control cell MUSTstop, ISI camera movement may continue.

Explanation/Justification:

A. Rod movement in cells with no fuel not required to stop, not considered a core alteration. Camera not considered a core alteration not required to stop.

B. Rod movement in cells with no fuel not required to stop, not considered a core alteration.

C. *Camera not considered a core alteration, not required to stop.*

D. Correct answer, Determine no SRMs operable, determine with no operable SRMs CORE ALTERATIONS must be suspended. Know definition of CORE ALTERATIONS and understand control rod movement with surrounding fuel cells is a core alteration. Camera movement is not a core alteration and can continue with no operable SRMs.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of refueling administrative requirements.**

K/A# **2.2.26** K/A Importance **2.5** Exam Level **RO**

References provided to Candidate **3.3.1.2** Technical References: **TS 3.3.1.2 & 1.1**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

72 A 22 year old operator is working in a radiation field under the following conditions:

The operators cumulative dose for the year is 940 mrem.
The job is in a 20 mrem/hr radiation area.
No dose extension has been or will be authorized.

Select the number of hours the operator may work in the radiation area without exceeding the administrative limit for the year?

A. 26 hrs

B. 53 hrs

C. 203 hrs

D. 153 hrs

Question Data
Answer: B 53 hrs

Explanation/Justification:

A. 1000 Admin limit (minus) 940 cum dose = 60 dose available (divided by) 20 dose rate = 26 hours
B. correct answer, 2000 Admin limit (minus) 940 cum dose = 1060 dose available (divided by) 20 dose rate = 53 hours
C. 5000 NRC (minus) 940 cum dose = 4060 dose available (divided by) 20 dose rate = 203 hours
D. 4000 Admin limit (minus) 940 cum dose = 1060 dose available (divided by) 20 dose rate = 153 hours

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Sys #	System	Category	KA Statement
		Radiological Controls	**Knowledge of 10 CFR 20 and related facility radiation control requirements.**

K/A# 2.3.1 K/A Importance 2.6 Exam Level **RO**
References provided to Candidate **None** Technical References: **NDAP-QA-0625 (6.2)**
Question Source: **Modified** **Clinton 1, 2001** Level Of Difficulty: (1-5) **3**
Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

73 - Unit 1 is in Mode 3.
- It is desired to de-inert the Unit 1 Primary Containment as soon as possible to permit containment access for Maintenance.

What flowpath is permitted for de-inerting of the Unit 1 Suppression Chamber?

A. Both Standby Gas Treatment trains in service, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

B. One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

C. Both Standby Gas Treatment trains in service, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703, SUPP CHMBR VENT OB ISO HV-15704, and SUPP CHMBR VENT BYPS OB ISO HV-15705.

D. One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704, and SUPP CHMBR VENT BYPS OB ISO HV-15705.

Question Data

Answer: B One Standby Gas Treatment train in service, the other Standby Gas Treatment Train operable, vent Suppression Pool via SUPP CHMBR VENT IB ISO HV-15703 and SUPP CHMBR VENT OB ISO HV-15704.

Explanation/Justification:

A. *Only one SBGT train I/S at a time.*
B. Correct answer.
C. Only one SBGT train I/S at a time, and either the 04 or 05 valve open, not both.
D. Either the 04 or 05 valve open, not both.

Sys #	System	Category	KA Statement
		Radiological Controls	**Knowledge of the process for performing a containment purge.**

K/A# **2.3.9** K/A Importance **2.5** Exam Level **RO**

References provided to Candidate **None** Technical References: **OP-173-001**

Question Source: **Modified** **Quad Cities 1, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.41**

74 EO-100-102, "RPV Control", directs the operator to reset the main generator lockout if RPV level can be maintained > -129 inches.

SELECT the correct basis for this step from the reasons listed below.

A. To allow the Stator Water Cooling Pumps to be restarted to provide cooling to the main generator.

B. To allow the Recirc Pumps to be restarted to establish forced reactor coolant circulation.

C. To prevent a trip of the Stator Water Cooling Pump.

D. To prevent a plant Auxiliary Bus load shed.

Question Data

Answer: D To prevent a plant Auxiliary Bus load shed.

Explanation/Justification:

A. No half scram signal received
B. No RRP runback signal exists
C. No Rod Block signal received
D. correct answer

Sys #	System	Category	KA Statement
		Emergency Procedures and Plan	**Knowledge of the specific bases for EOPs.**

K/A# **2.4.18** K/A Importance **2.7** Exam Level **RO**

References provided to Candidate **None** Technical References: **AR-103-001**

Question Source: **Modified** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.41**

75 The reactor is shut own with one loop of shutdown cooling in use and NO Recirculation Pumps running.

How would the Control Room operators respond if Reactor Vessel Level decreased from +50 inches to +4 inches?

A. Enter EO-100-102 RPV Control, maximize CRD and check the Shutdown Cooling Suction Inboard and Outboard Isolation Valves isolate, the operating RHR Pump will trip.

B. Enter ON-149-001, Loss of Shutdown Cooling, maximize RHR keepfill, check the Shutdown Cooling Suction Inboard and Outboard Isolation Valves isolate, the operating RHR Pump remain running.

C. Verify alarm response for RX WATER HI-LO LEVEL AR-101-B17, verify Shutdown Cooling continues unaffected.

D. Verify alarm response for HV-151-F006A/C AND HV-151-F007A OPEN DRAINS RX VESSEL AR-109-C09, verify the Shutdown Cooling Suction Inboard and Outboard Isolation Valves isolate, the operating RHR Pump will trip, HV-151-F015A will open, remaining RHR Pumps auto start.

Question Data

Answer: A Enter EO-100-102 RPV Control, maximize CRD and check the shutdown cooling suction inboard and outboard isolation valves isolate, the operating RHR pump will trip.

Explanation/Justification:

A. correct answer, running RHR pump will trip due to loss of suction path., Entry into EO required due to level less than +13 inches
B. No suction path running pump will trip.
C. No suction path, running pump will trip.
D. F015 valve auto opens and RHR pumps will auto start at -129.

Sys #	System	Category	KA Statement
		Emergency Procedures and Plan	**Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.**

K/A# 2.4.9 K/A Importance 3.3 Exam Level RO

References provided to Candidate **None** Technical References: **TM-OP-049**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

76 Unit 2 is operating at 90% power when a recirculation flow reduction event results in entry into Region 2 of the Power to Flow Map.

After the flow reduction event and core flow first reaches its lowest flowrate, which of the following instrumentation responses would you use as the Unit Supervisor to justify entry into the core oscillation Off Normal Procedure?

A. Peak-to-peak oscillations on RBM are 10% and growing larger.

B. Period meters are oscillating and short period alarms are received on a 10 to 20 second frequency.

C. Peak-to-peak oscillations on APRM's are 5% to 6% and their magnitude is growing larger.

D. Total Steam flow oscillations are 10 to 12% and their magnitude is growing larger

Question Data

Answer: C Peak to peak oscillations on APRM's are 5% to 6% and their magnitude is growing larger.

Explanation/Justification:

A. RBM not referenced in the off normal procedure.
B. No reference to period indication in the off normal procedure.
C. correct answer
D. Steam flow not referenced in the off normal procedure.

Sys #	System	Category	KA Statement
295001	**Partial or Complete Loss of Forced Core Flow Circulation**	**Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:**	**Neutron monitoring**

K/A# **295001.AA2.02** K/A Importance **3.2** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-278-002**

Question Source: **Modified** **Peach Bottom, 1995** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

77 Unit 1 is operating at 100% power when an Instrument Air line in the Turbine Building ruptures. The air compressors are unable to keep up with the loss of air and Instrument Air pressure is lowering.

What will the overall Reactor Pressure Vessel level control and pressure control strategy be for the loss of Instrument Air?

A. Main Steam Line drains for pressure control, Condensate Pumps for level control.

B. SRVs for pressure control, HPCI/RCIC for level control.

C. SRVs for pressure control, Maximize CRD for level control.

D. Bypass Valves for pressure control, HPCI/RCIC for level control.

Question Data

Answer: B SRVs for pressure control, HPCI/RCIC for level control.

Explanation/Justification:

A. Condenser is not available and no condensate line up is possible due to level control valves fail closed on a loss of air.

B. correct answer, Outboard MSIVs will go closed on a loss of air, therefore no steam for feedpumps or use of the main condenser for decay heat. Condensate will be unavailable due to no feedpath on a loss of air.

C. CRD flow control valves fail closed on a loss of air,

D. Condenser is not available for pressure control

Sys #	System	Category	KA Statement
295019	**Partial or Complete Loss of Instrument Air**	**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)**

K/A# **295019.AA2.02** K/A Importance **3.7** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-118-001, EO-100-102**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

78 Which of the following describes the consequences of a Design Basis Fuel Handling accident if the Refuel Floor High Exhaust Duct Rad Monitors and the Refuel Floor Wall Exhaust Duct rad monitors fail to function?

Any release as a result of this accident will:

A. not be processed by SGTS and may result in a Site Boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

B. not be processed, but will be monitored by Reactor Building Rad Monitors for conditions requiring entry into EO-100-105, Radioactivity Release Control and an Emergency Plan entry at the Site Area Emergency level.

C. will be processed by Zone 1 and Zone 2 HVAC, but still may result in a Site Boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

D. will be processed by Zone 1 and Zone 2 HVAC, but still may result in conditions requiring entry into EO-100-105, Radioactivity Release Control and an Emergency Plan entry at the Site Area Emergency level.

Question Data

Answer: A not be processed by SGTS and may result in a site boundary dose in excess of 25 rem whole body and an Emergency Plan entry at the General Emergency level.

Explanation/Justification:

A. Correct, inop rad monitors, SGTS won't start, release won't be processed, and release may exceed 10CFR100 limits at site boundary which is 25 Rem whole body and 300 Thyroid.

B. Incorrect, Reactor Building SPINGs will monitor release. Ventilation exhaust will be partially treated by the Zone III filtered exhaust.

C. Incorrect, Zone 3 independent of 1 and 2, therefore there would be no treatment of the exhaust by Zone I & II.

D. Incorrect, Zone 3 independent of 1 and 2, therefore there would be no treatment of the exhaust by Zone I & II.

Sys #	System	Category	KA Statement
295023	**Refueling Accidents**	**Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS:**	**Entry conditions of emergency plan**

K/A# **295023.AA2.05** K/A Importance Exam Level **SRO**

References provided to Candidate **None** Technical References: **Tech Spec B3.3.6.2**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

79 Unit 1 is operating at 100% power when a failure causes a Division I & II Core Spray initiation signal.

How will Drywell Temperature and Pressure respond to the inadvertent Core Spray initiation signal?

Assuming the initiation signal could not be reset, what actions would the Unit Supervisor direct for the above plant conditions?

A. Drywell Temperature and Pressure rise.
Scram imminent and scram from ON-100-101, Bypass DW Cooling Logic Isolations from EO-100-103.

B. Drywell Temperature rises, Pressure constant
Reduce power from GO-100-012, Bypass DW Cooling Logic Isolations from EO-100-103.

C. Drywell Temperature and Pressure rise.
Reduce power from GO-100-012, Vent Containment from ES-173-001

D. Drywell Temperature constant, Pressure constant
Shut down Core Spray pumps from OP-151-001

Question Data

Answer: A Drywell Temperature and Pressure rise.
Scram imminent and scram from ON-100-101, Bypass DW Cooling Logic Isolations from EO-100-103.

Explanation/Justification:

A. Correct answer, initial conditions show hot outside tempersatures to set up the loss of drywell cooling caused by the false Core Spray initiation. Initiation of Core Spray causes a Drywell Isolation and resultant loss of Drywell cooling. The loss of cooling causes drywell pressure and temperature to rise necessitating a scram and entry into EO-100-103 for containment control which initiates bypassing the containment isolation.

B. Pressure will rise. Can not bypass/use ES procedure unless direected from a EO procedure.

C. Can not bypass/use ES procedure unless direected from a EO procedure.

D. Temperature and pressure will increase.

Sys #	System	Category	KA Statement
295024	**High Drywell Pressure**	**Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:**	**Drywell temperature**

K/A# **295024.EA2.02** K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-000-102 or 103**

Question Source: **Modified** **Susquehanna, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

80 Unit 1 is operating at 100% power when a MSIV isolation occurs causing the following:

- Partial scram with one-quarter of the control rods not fully inserted.
- Reactor pressure peaked at 1135 psig.

Which of the following correctly lists the automatic actions that should occur to control the pressure transient and what procedure(s) will be used for reactor vessel pressure control?

A. Reactor Recirc Pump trip, SRV operation. EO-100-113

B. Main Turbine Bypass valves operate, SRV operation. EO-100-113

C. Reactor Recirc Pump trip, ARI. EO-100-102, EO-100-113

D. ARI, SRV operation. EO-100-102, EO-100-113

Question Data

Answer: A Reactor Recirc Pump trip, SRV operation. EO-100-113

Explanation/Justification:

A. correct answer, Recirc pump ATWS trip at 1123, SRV ops at 1126. EO-100-113 for ATWS.
B. Bypass valves not available with MSIVs closed.
C. EO-100-102 entered but exited due to more than one rod out, thus not used for pressure control.
D. EO-100-102 entered but exited due to more than one rod out, thus not used for pressure control.

Sys #	System	Category	KA Statement
295025	**High Reactor Pressure**	**GENERIC**	**Ability to perform specific system and integrated plant procedures during different modes of plant operation**

K/A# 295025.2.1.23 K/A Importance 4.0 Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **AR-103-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

81 Unit 1 experienced a 100% power load reject and a complete Loss of Offsite Power:

- Reactor scram	
- RPV Pressure	340 psig
- RPV Level	-10 inches
- Drywell Pressure	+14 psig
- Drywell Temperature	255 degrees F
- Suppression Pool Level	16 feet
- Suppression Chamber Pressure	+16 psig
- Suppression Pool Temperature	270 degrees F

For the given plant conditions, what systems are available for Reactor Vessel Level control?

A. RCIC, SLC and CRD

B. RHR Service Water, Core Spray, CRD

C. Condensate Pumps, RHR, and SLC

D. Core Spray, HPCI, and CRD

Question Data
Answer: A RCIC, SLC and CRD

Explanation/Justification:
A. correct answer, RCIC can be operated using CST suction, SLC and CRD have power available from Diesel Generators.
B. RHR SW and Core Spray do not have the required discharge pressure at this time
C. No power to the condensate pumps, RHR does not have sufficient discharge pressure.
D. HPCI shouldn't be run at >140 deg F due to lube oil cooling and should be isolated at 17' suppression pool level.

Sys #	System	Category	KA Statement
295026	**Suppression Pool High Water Temp**	**GENERIC**	**Knowledge of abnormal condition procedures**

K/A# **295026.2.4.11** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-100-009**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

SSES LOC 19 NRC Exam

82 A LOCA has occurred, RPV level has slowly trended down. The following conditions exist in the plant:

Reactor Pressure	50 psig
Wide Range Level	- 146 inches
Fuel Zone Level	- 93 inches
Upset Range Level	0 inches
Shutdown Range Level	0 inches
Narrow Range Level	0 inches
Drywell Pressure	+ 5.2 psig
Drywell Temperature	200 degrees F

004 BJB 8/19/03

As the Unit Supervisor using ON-145~~-001,~~ which of the following Reactor Level Instruments would you instruct the operators to use as water level lowers?

A. Shutdown Range

B. Fuel Zone

C. Upset Range

D. Wide Range

Question Data

Answer: D Wide Range

Explanation/Justification:

A. 0-500, calibrated cold, below useable range

B. Fuel Zone is not on scale for stated conditions reviewing the graph from ON-145-004

C. 0 - 180 calibrated hot.

D. correct answer, using the ON Wide Range level can be extended to -147 for the stated conditions, Fuel Zone is not on scale for stated conditions.

Sys #	System	Category	KA Statement
295028	**High Drywell Temperature**	**Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:**	**Reactor water level**

K/A# **295028.EA2.03** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **ON-145-004** Technical References: **ON-145-004**

Question Source: **Modified** **Grand Gulf 1, 2000** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

83 An ATWS from 100% power has occurred on Unit 1. The following conditions now exist:

RPV level band:	-60 inches to -161 inches
Reactor Power:	25 on IRMs range 6
SLC Tank level	5%

Based on the above conditions, what should be the next direction/instruction given to the shift crew?

A. Restore and maintain RPV level -60 to -110 inches.

B. Commence a controlled cooldown of the reactor vessel per EO-100-102.

C. "Attention Crew update, entering EO-100-112."

D. Restore and maintain RPV level in the normal band +13 to +54 inches.

Question Data

Answer: D restore and maintain RPV level in the normal band, +13 to +54 inches

Explanation/Justification:

A. with HSBW added, level is restored to +13 to +54 inches.

B. Rx is still critical can't commence a cooldown. Cooldown as per EO-100-113, not 102 board.

C. Rapid depressurization not directed at present level and steady or holding

D. correct answer, SLC tk level is less than 2800 gallons therefore HSBW has been added or boron injection has occurred for >24 minutes therefore the HSBW has been added and level should be restored to +13-54

Sys #	System	Category	KA Statement
295031	**Reactor Low Water Level**	**GENERIC**	**Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations**

K/A# 295031.2.4.22 K/A Importance **4.0** Exam Level **SRO**

References provided to Candidate **Emergency Operating Procedures** Technical References: **EO-100-113**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

84 A Unit 1 startup was in progress, with Main Condenser vacuum at 1.5 inches Hg Absolute and Reactor power at 22%.

Main Condenser vacuum has decayed to 10.0 inches Hg Absolute and is presently steady.

How will reactor power respond to the lower vacuum and what procedure(s) must be implemented to address these plant conditions?

A. Reactor Power will be 0% due to the turbine trip and reactor scram. ON-100-101, Scram and EO-100-102 RPV Control

B. Reactor power will be greater than 22% due to a loss of feed water heating. ON-143-001, Loss of Main Condenser Vacuum, and ON-100-101, Scram.

C. Reactor power will be 0% due to the Reactor Feed Pumps tripped, and reactor scram. ON-100-101, Scram and EO-100-102 RPV Control.

D. Reactor Power will be less than 22% due to the decrease in condensate subcooling. ON-100-113 Level/Power Control.

Question Data

Answer: B Reactor power will be greater than 22% due to a loss of feed water heating. ON-143-001, Loss of Main Condenser Vacuum, and ON-100-101, Scram.

Explanation/Justification:

A. No scram on turbine trip, bypassed at power <30%.

B. Correct answer, Main Turbine trips at 8.2" Hg Absolute (21.7" Hg Vacuum) which will cause a loss of feedwater heating with the loss of extraction steam causing reactor power to rise. The reactor scram on a turbine trip is bypassed at <30 % power so no reactor scram. The loss of condenser vacuum Off Normal would be implemented and the off normal implements the Scram procedure for scram imminent actions.

C. No loss of RFPs on loss of vacuum until 17.4" Hg Vacuum.

D. Power will change due to Turb trip but no scram required, bypassed at power <30 %.

Sys #	System	Category	KA Statement
295002	**Loss of Main Condenser Vacuum**	**Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM:**	**Reactor power: Plant-Specific**

K/A# **295002.AA2.02** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **ON-143-001**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

85 Given the following conditions on Unit 1:

- Reactor power has been lowered to 95% in preparation for Turbine Control Valve testing
- Prior to starting the test, the PCOM reports APRM reactor power is rising
- Power peaks and stabilizes at 100%
- No alarms are received
- After investigation, the PCOP discovers HPCI is running and injecting
- All other equipment and instruments are operating as designed

HPCI initiation caused by Relay Room cabinet door jarring relays.

Which of the following is the required Unit Supervisor direction regarding reactor power and the reportablity requirements for these conditions?

The US shall direct a Recirc Flow reduction to:

A. less than 75% power and ensure an 8 hour reportability call is made.

B. less than or equal to 95% power and 4 hour reportability.

C. less than 75% power and ensure a 4 hour reportability call is made.

D. less than or equal to 95% power and ensure a 8 hour reportability call is made

Question Data

Answer: B less than or equal to 95% power and 4 hour reportability.

Explanation/Justification:

A. power reduction to 75% is required for a loss of feedwater heating.
B. correct answer, power is returned to the original starting power and notification is made due to ECCS injecting per 10CFR50.72(b)(2)(iv)(A)-4 Hour ENS..
C. power reduction to 75% is required for a loss of feedwater heating. Incorrect reportability time.
D. Incorrect reportability time.

Sys #	System	Category	KA Statement
295014	**Inadvertent Reactivity Addition**	**Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION:**	**Reactor power**

K/A# **295014.AA2.01** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **NDAP-QA-0720 Attachment G** Technical References: **ON-156-001 & NDAP-QA-0720**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

86 A Reactor Scram occurred. There are still 20 rods at Position 48. The following sequence of events takes place:

- Operators maximize CRD.
- Scram and ARI are reset.
- A break in the instrument air header causes a loss of instrument air to the scram air header.

Which of the following methods should be pursued to insert control rods in this new condition?

A. Direct individually scramming of Control Rods with SRI Switches locally.

B. Direct closing the Charging Valve and attempting to manually drive Control Rods.

C. Reset the scram and attempt an additional manual scram.

D. Direct de-energizing scram solenoids by removing RPS fuses.

Question Data

Answer: B Direct closing the charging valve and attempting to manually drive control rods.

Explanation/Justification:

A. Without air, scram inlet and outlet valves should already be open.
B. correct answer,
C. Without air, scram cannot be reset since the discharge vent and drain valves remain closed.
D. Scram inlet/outlet valves already open on loss of air.

Sys #	System	Category	KA Statement
295015	**Incomplete SCRAM**	**GENERIC**	**Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation**

K/A# 295015.2.1.7 K/A Importance 4.4 Exam Level **SRO**

References provided to Candidate **EOP Flowcharts** Technical References: **EO-100-113**

Question Source: **Bank** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

87 IF:

- A loss of both CRD Pumps has occurred during a reactor startup
- Reactor pressure at 500 psig.
- NPO dispatched to determine accumulator pressure for Control Rod 30-35.
- Control Rod 30-35 is at Position 12.
- A Crew Brief is in progress to review event and actions to take.

As the Unit Supervisor which of the following actions are required if during the crew briefing, the NPO reports HCU 30-35 pressure at 920 psig?

A. Continue the brief, direct the PCOM to check for the second accumulator alarm, then place the Mode Switch to Shutdown.

B. Suspend the brief, direct the PCOM to insert Control Rod 30-35 one notch.

C. Continue the brief, within 20 minutes direct the PCOM to restart a pump and insert Control Rod 30-35 one notch.

D. Suspend the brief, direct the PCOM to immediately place the Mode Switch to Shutdown.

Question Data

Answer: D Suspend the brief, direct the PCOM to immediately place the Mode Switch to Shutdown.

Explanation/Justification:

A. No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
B. No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
C. No CRD pumps, Rx pressure less than 900, accumulator alarm, Mode Switch to S/D.
D. correct answer, If reactor steam dome pressure < 900 psig and one or more scram accumulators are inoperable, PLACE Reactor Mode Switch in SHUTDOWN position.

Sys #	System	Category	KA Statement
295022	**Loss of CRD Pumps**	**Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:**	**Accumulator pressure**

K/A# 295022.AA2.01 K/A Importance 3.6 Exam Level SRO

References provided to Candidate **None** Technical References: **ON-155-007**

Question Source: **Modified** **Nine Mile Point 1, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

88 Given the following conditions with Unit 1 in Mode 4 and Unit 2 at 100% power

Division 1 LOOP/LOCA testing is in progress on Unit 1.
As part of the test, the A & C Diesel Generators start and load to their respective Unit 1 busses.
ECCS responses are as follows:

- 1A RHR Pump	Does NOT start
- 1C RHR Pump	Starts at 10 seconds
- 1A Core Spray Pump	Starts at 10 seconds
- 1C Core Spray Pump	Starts at 10 seconds

Which of the following Tech Spec actions should be taken?

A. Declare 1A RHR Pump Inoperable, Unit 1 enters Tech Spec 3.5.2.
Declare C DG Inoperable, Unit 1 and 2 enter Tech spec 3.8.2.

B. Declare 1A RHR Pump Inoperable, no Tech Spec entry required.
Declare C DG Inoperable, Unit 2 enters Tech Spec 3.8.1.
C DG can be returned to Operable if 1C RHR Pump Breaker DC Knife Switch opened.

C. Declare 1A RHR Pump Inoperable, no Tech Spec entry required.
Declare 1C RHR Pump Inoperable, no Tech Spec entry required.
C DG remains Operable.

D. Declare 1A and 1C RHR Pumps Inoperable, Unit 1 enters Tech Spec 3.5.2.
Declare A & C DGs Inoperable, Unit 2 enters Tech Spec 3.8.1.
A DG can be returned to Operable if 1A RHR Pump breaker DC Knife Switch opened.

Question Data

Answer: B Declare 1A RHR Pump Inoperable, no Tech Spec entry required
Declare C DG Inoperable, Unit 2 enters Tech Spec 3.8.1
C DG can be returned to Operable if 1C RHR Pump breaker DC Knife Switch opened.

Explanation/Justification:

A. 3.5.2 U-1 S/D ECCS is not applicable with one pump out of service. 3.8.2 S/D Electrical is not applicable for unit 2.

B. correct answer, With Unit 1 in Mode 4 the 1A RHR pump can be declared out of service with the other pump in the loop still operable and no entry into Tech Specs. Surveillance requirements for Unit 2 D/G operability in Mode 1 requires DG auto-starts from standby condition and: energizes permanently connected loads in < 10 seconds, energizes auto-connected shutdown loads through individual load timers. With the 1C RHR pump DC knife switch open the pump not loading in the proper time does not inop the D/G for unit 2.

C. 1C RHR pump is not inoperable.

D. 3.5.2 U-1 S/D ECCS is not applicable with one pump out of service.

Sys #	System	Category	KA Statement
203000	**RHR/LPCI: Injection Mode (Plant Specific)**	**Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Emergency generator failure**

K/A# **203000.A2.06** K/A Importance **3.9** Exam Level **SRO**

References provided to Candidate **TS 3.5.2, 3.8.1 & 3.8.2** Technical References: **TM-OP-049**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

89 Given the following conditions:

- Unit 1 is in Mode 4 with Shutdown Cooling in service utilizing the 'A' Residual Heat Removal (RHR) Pump.
- Electrical Maintenance has reported that one of the B RPS MG Set EPA Breakers is running hotter than acceptable.
- They recommend 'B' RPS be transferred to the Alternate Power Supply.

Which of the following will prevent a loss of Shutdown Cooling (SDC) and addresses the operability issues of opening breakers?

Shutdown Cooling isolation and 'A' RHR Pump trip will be prevented by:

A. opening the breaker supplying power to SDC Suction Outboard Isolation Valve (HV-151-F008) and entering LCO 3.6.1.3, Condition A.

B. opening the breakers supplying power to SDC Suction Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and entering LCO 3.6.1.3, Condition B.

C. opening the breakers supplying power to SDC Suction Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and entering LCO 3.4.9, Condition A.

D. opening the breaker supplying power to SDC Suction Outboard Isolation Valve (HV-151-F008) and NO LCO is required due to the plant being in Mode 4.

Question Data

Answer: B opening the breakers supplying power to SDC Suction Inboard and Outboard Isolation Valves (HV-151-F009 and F008) and entering LCO 3.6.1.3, Condition B.

Explanation/Justification:

A. Incorrect, loss of B RPS cause both F008 and F009 to close.
B. Correct, allowed by procedure
C. Incorrect, SDC is still operable, the isolation capability isn't.
D. incorrect, loss of B RPS causes both F008 and F009 to close, must take LCO

Sys #	System	Category	KA Statement
205000	**Shutdown Cooling System (RHR Shutdown Cooling Mode)**	**Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM/MODE including:**	**Pump trips**

K/A# **205000.A3.02** K/A Importance **3.2** Exam Level **SRO**

References provided to Candidate **TS 3.4.9 & 3.6.1** Technical References: **OP-158-001 Att B**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

90 Given the following conditions with Unit 1 at 100% power:

- Core Spray Loop 'A' Header Break Detection High Differential Pressure alarm has just been received
- The Reactor Building NPO reports PDIS-E21-1N004A is reading -3.4 psid on Panel 1C010.

Which of the following actions are required for these conditions?

A. Declare Core Spray 'A' header D/P instrumentation channel inoperable. Restore to Operable status within 72 hours.

B. Declare Core Spray Loop 'A' Inoperable. Restore to Operable status within 72 hrs.

C. Declare Core Spray Loop 'A' operable. Write an AR to document the out-of-specification differential pressure condition.

D. Declare Core Spray Loop 'A' operable. Write a tracking LCO to document the system's ability to inject inside the vessel but not spray above the core.

Question Data

Answer: A Declare Core Spray A header d/p instrumentation channel inoperable. Restore to Operable status within 72 hours.

Explanation/Justification:

A. Correct.
B. Incorrect. CS still operable, d/p still good, bad alarm
C. Incorrect. D/P still good, AR should be for incorrect alarm received
D. Incorrect. CS still capable of performing its intended function

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Sys #	System	Category	KA Statement
209001	**Low Pressure Core Spray System**	**Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:**	**Core spray line break**

K/A# **209001.A2.05** K/A Importance **3.6** Exam Level **SRO**

References provided to Candidate **TRO 3.5.2** Technical References: **TRO 3.5.2, Pages 3.5-3 through 3.5-5**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

91 Given the following conditions with Unit 2 in Mode 1:

- Surveillance SO-024-001, Monthly Diesel Generator Operability Test, is being performed for the A DG to the 2A201 bus.
- The 2A201 Bus is currently loaded at 800 kW
- The A DG is operating at the SO hold point of 1000 kW.
- The Supply Breaker to 2A201 from Transformer 0X201 trips due to a breaker problem.

Which of the following describes the expected electric plant response and the Unit Supervisor directed actions for these conditions?

The 'A' Diesel Generator Output Breaker:

A. trips and the US should direct resetting the 2A201 Bus lockouts and power restoration from its alternate power source.

B. trips and the US should direct verification of 2A201 automatically re-energizing from its alternate power source.

C. does NOT trip and the US should direct restoration of normal bus voltage and frequency parameters.

D. does NOT trip and the US should direct an immediate trip of the Diesel Generator.

Question Data

Answer: C does NOT trip and the US should direct restoration of normal bus voltage and frequency parameters.

Explanation/Justification:

A. Incorrect, breaker does not trip, no lockouts trip, bus remains energized

B. Incorrect, breaker does not trip, bus remains energized

C. Correct, a loss of 200 KW will show up as changes in frequency and voltage, return parameters to normal bands.

D. Incorrect, DG can handle this transient without breaker or engine tripping, no reason to direct trip

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Sys #	System	Category	KA Statement
264000	**Emergency Generators (Diesel/Jet)**	**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:**	**Loss of A.C. power**

K/A# **264000.A2.09** K/A Importance **4.1** Exam Level **SRO**

References provided to Candidate **None** Technical References: **SO-024-001, Electrical Theory and application**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

92 Unit 1 is at 100% power with no LCOs entered.

Electrical Maintenance has submitted a work package to overhaul the HV-151-F006A Shutdown Cooling Suction Valve motor actuator.

The work package requires the HV-151-F004A to be closed to allow manual operation of the HV-151-F006A to set the valve limit switches.

What Limiting Conditions of Operation will be in effect?

A. No LCO required

B. LCO 3.4.9, LCO 3.5.1, LCO 3.6.1.3

C. LCO 3.5.1

D. LCO 3.5.1, LCO 3.6.2.3, LCO 3.6.2.4

Question Data

Answer: C LCO 3.5.1

Explanation/Justification:

A. 'A' RHR pump is required to be inoperable.

B. SDC not required at rated conditions.

C. correct answer, to open the 06, the 04 suction valve must be closed, taking 'A' RHR pump out of service.

D. ECCS LCO required for RHR pp out of service.

Sys #	System	Category	KA Statement
219000	**RHR/LPCI: Torus/Pool Cooling Mode**	**GENERIC**	**Ability to analyze the affect of maintenance activities on LCO status**

K/A# **219000.2.2.24** K/A Importance **3.8** Exam Level **SRO**

References provided to Candidate **TS 3.4, 3.5 & 3.6** Technical References: **TS 3.5.1**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.41**

93 With Unit 2 in Mode 5 and the Refueling Platform approaching the Unit 2 reactor vessel, which of the following will initiate a rod withdraw block for any selected rod?

A. Grapple NOT engaged.

B. Any Hoist extended.

C. Any rod NOT fully inserted.

D. Any Hoist Loaded.

Question Data

Answer: D Any Hoist Loaded.

Explanation/Justification:

A. grapple not engaged will not cause a rod block.
B. no interlock for hoist extended, everything uses load cells.
C. 1 rod allowed full out.
D. Correct answer,

Sys #	System	Category	KA Statement
234000	**Fuel Handling Equipment**	**Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including:**	**Interlock operation**

K/A# **234000.A3.02** K/A Importance **3.7** Exam Level **SRO**

References provided to Candidate **None** Technical References: **TM-OP-56**

Question Source: **Modified** **Nine Mile Point 1, 1996** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

94 An accident occurred on Unit 2 while at 100% power, resulting in fuel damage.

The reactor scram fails.

The following plant conditions exist:

- Reactor power	18%
- Reactor pressure	940 psig
- RPV water level	-100 inches
- Main Steam Line B	Inboard & Outboard MSIVs failed open
- Main Turbine	Tripped
- Site boundary release (Adult Thyroid)	4.90 Rem (rising)

Given the above conditions, which of the following actions are required?

A. Use the SRVs to maintain reactor pressure less than 965 psig.

B. Use Main Turbine BPVs to commence a reactor cooldown at less than a 90F/Hr rate.

C. Use Main Turbine BPVs to commence a reactor cooldown at greater than a 90 degrees F/Hr rate.

D. Perform Emergency RPV Depressurization.

Question Data

Answer: D Perform Emergency RPV Depressurization.

Explanation/Justification:

A. EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

B. EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

C. EO-200-105 rad release EO, leads to EO-200-102 RPV level control but back out to the level/pwer control EO therefore there is no cooldown allowed.

D. correct answer, approaching 5 Rem for Adult Thyroid and MSIVs failing to close is a "Primary system" discharging. The ATWS is being controlled by EO-200-113 with level being maintained at -100" as required by the procedure.

Sys #	System	Category	KA Statement
		Conduct of Operations	**Ability to execute procedure steps.**

K/A# **2.1.20** K/A Importance **4.2** Exam Level **SRO**

References provided to Candidate **EOP Flow Charts** Technical References: **EOP Flow charts**

Question Source: **Modified** **Fermi 2 2, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

95 In accordance with Unit 1 Tech Specs and ON-183-001, Stuck Open Safety Relief Valve, the Reactor Mode Switch was placed in Shutdown at 108 degrees F due to a stuck open SRV.

Post-scram, the Suppression Pool reached a peak of 114 degrees F before Suppression Pool Cooling was able to begin removing heat. The reactor was NOT required to be placed in Mode 4.

Which of the following describes the restrictions on the ensuing reactor startup? Assume the SRV has been repaired.

Suppression Pool temperature must be less than or equal to:

A. 105 degrees F prior to placing the Reactor Mode Switch in Startup/Standby.

B. 90 degrees F prior to placing the Reactor Mode Switch in Startup/Standby.

C. 90 degrees F prior to exceeding 1% power.

D. 105 degrees F prior to exceeding 1% power.

Question Data

Answer: C 90 degrees F prior to exceeding 1% power.

Explanation/Justification:

A. Incorrect, 105 limit only applies >1% power with testing in progress

B. Incorrect, no requirement for this since the reactor remains in one of the three modes for which the LCOs apply at all times

C. Correct, in Modes 1, 2 and 3 this limit applies once >1% power.

D. Incorrect, 105 limit only applies >1% power with testing in progress.

Sys #	System	Category	KA Statement
		Conduct of Operations	**Knowledge of less than one hour technical specification action statements for systems.**

K/A# 2.1.11 K/A Importance **3.8** Exam Level **SRO**

References provided to Candidate **TS 3.6.2** Technical References: **TS 3.6.2.1**

Question Source: **New** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**

96 Which of the following are considered Unreviewed Safety Questions?

1. Emergency actions that depart from T.S. are needed to protect the public health and safety.
2. The possibility of an accident exists that has not been evaluated by the FSAR.
3. The consequence of a malfunction of equipment evaluated by the FSAR is increased.
4. The margin of safety as defined in T.S. is reduced.
5. An emergency event that can not be classified by the Emergency Plan.

A. 2, 3, 4

B. 1, 2, 3

C. 3, 4, 5

D. 2, 3, 5

Question Data

Answer: A 2, 3, 4

Explanation/Justification:

A. correct answer, Not USQ - Emergency actions that depart from T.S. are needed to protect the public health and safety.
Is USQ - The possibility of an accident exists that has not been evaluated by the FSAR
Is USQ - The consequence of a malfunction of equipment evaluated by the FSAR is increased.
Is USQ - The margin of safety as defined in T.S. is reduced.
Not USQ - An emergency event that can not be classified by the Emergency plan.

B. 1, not USQ

C. 5, not USQ

D. 5, not USQ

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of the process for determining if the proposed change, test, or experiment involves an unreviewed safety question.**

K/A# **2.2.8** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **NDAP-QA-0726**

Question Source: **Modified** **Columbia Gen Sta 2, 1999** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

97 Which of the following is the bases for the Technical Specifications, Minimum Suppression Chamber Water Volume in Modes 1, 2, and 3?

A. Ensures a sufficient supply of water is available, with the Minimum CST Volume in the event of a LOCA to permit recirculation cooling flow to the core.

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, with the Minimum CST Volume, Long-Term Cooling is available for the Design Basis Accident.

Question Data

Answer: 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.

Explanation/Justification:

A. CST is not part of basis.
B. HPCI & RCIC are included in the basis.
C. correct answer, 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. CST is not part of basis.

Sys #	System	Category	KA Statement
		Equipment Control	**Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.**

K/A# 2.2.25 K/A Importance 3.7 Exam Level **SRO**

References provided to Candidate **None** Technical References: **TS Basis - B 3.6.2.2**

Question Source: **Modified** **Fermi 2, 1998** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Fundamental** 10 CFR Part 55 Content: **55.43**

98 Maintenance must be performed in the Unit 1 Reactor Water Cleanup (RWCU) Backwash Receiving Tank Room (Room 1-509) to support a modification.

Per NDAP-QA-0626, Radiologically Controlled Area Access and Radiation Work Permit (RWP) System, which of the following actions is required BEFORE the Unit Supervisor may allow work to begin in the room to comply with the ALARA BLOCKING principle?

A. Flush, then drain the Backwash Receiving Tank and maintain the tank empty.

B. Backwash the RWCU filters, then drain the filters and maintain the filters empty.

C. Flush and drain, then fill the Backwash Receiving Tank and maintain the tank full.

D. Backwash the RWCU filters, then fill the tank and maintain the tank full.

Question Data

Answer: C Flush and drain, then fill the Backwash Receiving Tank and maintain the tank full.

Explanation/Justification:

A. The tanks must also be filled with water to act as shielding.

B. This would not help the radiation levels in the Backwash Receiving Tank Rooms and may make it worse since the filters are backwashed to the room.

C. correct answer, NDAP-QA-0323, Standard Blocking Practices requires the tanks be flushed drained and filled before entry into the rooms.

D. This would not help the radiation levels in the Backwash Receiving Tank Rooms and may make it worse since the filters are backwashed to the room.

Sys #	System	Category	KA Statement
		Radiological Controls	**Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.**

K/A# **2.3.10** K/A Importance **3.3** Exam Level **SRO**

References provided to Candidate **None** Technical References: **NDAP-QA-0626, Sect. 6.2.5**

Question Source: **Modified** **Susquehanna, 2001** Level Of Difficulty: (1-5) **3**

Question Cognitive Level: **Memory** 10 CFR Part 55 Content: **55.43**

99 Maintenance has just reported to the Control Room that a 55-gallon drum of lube oil leaked into the River Water Makeup intake bay overnight.

Using the attachments from NDAP-QA-0720, STATION REPORT MATRIX AND REPORTABILITY EVALUATION GUIDANCE, determine ALL of the Offsite Agencies that must be notified after PA DEP is notified.

A. LCEMA, Coast Guard, NRC

B. CCEMA, Coast Guard, NRC

C. LCEMA, Coast Guard, PEMA

D. LCEMA, CCEMA, PEMA

Question Data

Answer: A LCEMA, Coast Guard, NRC

Explanation/Justification:

A. correct answer, PA DEP, LCEMA, and Coast Guard must be notified due to a petroleum product being discharged to a waterway. The NRC must be notified anytime an offsite agency is notified.

B. NRC must be notified.

C. Correct for a Comprehensive Environmental Response, Compensation, and Liability Act release which this spill is NOT.

D. Columbia county not notified for this spill not PA Emergency Management Agency.

Sys #	System	Category	KA Statement
		Emergency Procedures and Plan	**Knowledge of which events related to system operations/status should be reported to outside agencies.**

K/A# 2.4.30 K/A Importance 3.6 Exam Level **SRO**

References provided to Candidate **NDAP-QA-0720 Att Q & T** Technical References: **NDAP-QA-0720**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Comprehension** 10 CFR Part 55 Content: **55.43**

100 The Mode Switch has been placed in Shutdown.
Reactor Power is 15%.
Reactor Water Level is being lowering at 1 inch per minute.
Drywell pressure is 0.75 psig and slowly rising.

The Unit Supervisor is about to give direction from EO-100-113, LEVEL/POWER CONTROL to inhibit ADS and bypass MSIV and CIG interlocks.

Why is the Unit Supervisor giving this direction?

A. Current level control range is near the auto initiation of ADS which could allow uncontrolled low pressure injection and main condenser would be used as a heat sink for as long as possible.

B. Initiation of ADS and/or closure of the MSIVs would cause large pressure changes. GE calculations indicate swings in pressure cause large power oscillations.

C. Initiation of ADS would cause large pressure changes with resulting power oscillations. MSIVs must remain open for use of the main condenser as a heat sink.

D. MSIVs must remain open for use of the main condenser as a heat sink. CIG must remain available for later use of ADS Valves.

Question Data

Answer: A Current level control range is near the auto initiation of ADS which could allow uncontrolled low pressure injection and main condenser would be used as a heat sink for as long as possible.

Explanation/Justification:

A. correct answer, Prevent low pressure unborated injection and prevent MSIVs from closing due to a loss of pneumatic supply.
B. ADS would depressurize the RPV.
C. ADS would depressurize the RPV.
D. CIG not required, 2200# bottles supply header.

Sys #	System	Category	KA Statement
		Emergency Procedures and Plan	**Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.**

K/A# 2.4.49 K/A Importance 4.0 Exam Level **SRO**

References provided to Candidate **None** Technical References: **EO-000-113**

Question Source: **New** Level Of Difficulty: (1-5) **4**

Question Cognitive Level: **Analysis** 10 CFR Part 55 Content: **55.43**