U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information	
Name:	
Date:	Facility/Unit:
Region: I ☐ II ☐ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____

Applicant's Signature

Results		
Examination Value	75	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information

Name:	
Date:	Facility/Unit:
Region: I ☐ II ☐ III ☐ IV ☒	Reactor Type: W ☐ CE ☐ BW ☐ GE ☒
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 1	21328	01	02/19/2009	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	5	Multiple Choice	BANK

Topic Area	Description
Inst & Control	COR0022102, RPS Consequences of an NI failure

Related Lessons	
COR0022102	REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001090B	Predict the consequences a malfunction of the following would have on the RPS system: Nuclear instrumentation

Related References	
4.1.3	Average Power Range Monitoring System
4.5	Reactor Protection/Alternate Rod Insertion Systems
10CFR55.41(b)6	

Related Skills (K/A)	
215005.K1.01	Knowledge of the physical connections and/or cause- effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) RPS (4.0/4.0)

QUESTION: RO 1

A plant startup is in progress. The following conditions exist:

- Reactor is critical on range 6 of the IRMs
- Both IRM E and G's internal power supplies fail (no voltage).
- Only IRM G is bypassed

The startup is on hold while I&C initiates repair. Due to a mis-communication the APRM B Mode Switch is placed to STANDBY.

Which one of the following result from the APRM B mode switch being taken out of operate?

a. Full reactor scram occurs.

b. Half scram on the "A" RPS Channel only.

c. Half scram on the "B" RPS Channel only.

d. Only annunciator 9-5-2/B-1, NEUTRON MONITORING TRIP re-alarms.

ANSWER: RO 1
a. Full reactor scram occurs

Explanation:

A full reactor scram occurs because there is one unbypassed inoperative IRM in RPS Channel A (IRM E) and placing APRM B (RPS Channel B) out of OPERATE will cause it to become inoperative. This creates a situation where there is an inoperative nuclear instrument in each RPS Channel.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 2	15120	00	07/12/2001	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Electrical	COR0020702, DC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION ELE7000006 DC Electrical Distribution

Related Objectives
COR0020702001060D Describe the interrelationship between the DC Electrical Distribution System and the following:: Battery ventilation

Related References
COR0020702 DC Electrical Distribution COR0020701 DC Electrical Distribution 10CFR55.41(b)4

Related Skills (K/A)
263000.K1.03 Knowledge of the physical connections and/or cause- effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Battery ventilation (2.6/2.8)

QUESTION: RO 2

The plant has experienced a loss of 250 VDC Battery 1A Room Exhaust Fans which can not be recovered. Which of the following parameters is the major operational concern?

a. Temperature

b. Humidity

c. Hydrogen

d. Oxygen

ANSWER: RO 2

c. Hydrogen

Explanation:

The function as described in the USAR of the Battery room exhaust fans:

The battery exhaust system provides for dilution and removal of potentially flammable concentrations of hydrogen in the battery rooms.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 3	16402	02	11/09/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	COR0021802, RCIC - How does RCIC respond to a loss of AA-2

Related Lessons	
COR0021802	OPS Reactor Core Isolation Cooling

Related Objectives	
COR0021802001060C	State the electrical power supply to the following RCIC Items: Flow controller
COR0021802001100B	Predict the consequences of the following on the RCIC system: AC and/or DC Electrical power failure
10CFR55.41(b)7	

Related References	
2.3_9-4-1	Panel 9-4 - Annunciator 9-4-1
GE Drawing 791E264, Elementary Control Diagram	
10CFR55.41(b)7	

Related Skills (K/A)	
217000.K2.02	Knowledge of electrical power supplies to the following: (CFR: 41.7) RCIC initiation signals (logic) (2.8*/2.9*)

QUESTION: RO 3

The plant is at 100% power when 9-4-1/A-3, RCIC LOGIC POWER FAILURE, alarms. After investigation it is determined that 125V DC Panel AA2 is de-energized.

How would RCIC respond to a valid RCIC initiation signal?

a. RCIC automatically starts and injects

b. RCIC automatically starts but the injection valve does not inject.
RCIC can be manually started and cannot be made to inject at rated flow from the control room.

c. The RCIC system aligns for injection except RCIC-MO-131 remains closed.
RCIC can be manually started and inject at rated flow from the control room.

d. No RCIC components realign.

ANSWER: RO 3

c.. The RCIC system aligns for injection except RCIC-MO-131 remains closed.
RCIC can be manually started and inject at rated flow from the control room.

Explanation:

RCIC will not auto start since power was lost to relay which opens MO-131 RCIC can be manually aligned only.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 4	24762	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	4	Multiple Choice	NEW

Topic Area	Description

Related Lessons
COR0021902 Reactor Equipment Cooling

Related Objectives	
COR0021902001040A	Describe the REC design features and/or interlocks that provide for the following: Repositioning of REC Supply Valves to components
COR0021902001070D	Predict the consequences a malfunction of the following would have on the REC system: Loss of Normal AC power

Related References
5.2REC 5.3EMPWR 10CFR55.41(b)7

Related Skills (K/A)	
400000.K2.02	Knowledge of electrical power supplies to the following: (CFR: 41.7) CCW valves (2.9/2.9)

QUESTION: RO 4

The plant is operating at near rated power when a loss of offsite power occurs with a simultaneous loss-of-coolant accident. EDG 'A' fails to start.

One minute later and prior to any operator action to restore power what REC system valves, if any, are powered from the emergency diesel generator?

REC-MO-702, the REC Drywell supply valve
REC-MO-704, the REC Drywell return valve
REC-MO-711, North Critical Loop Supply
REC-MO-713, REC HX B Outlet Valve
REC-MO-714, South Critical Loop Supply

a. NO REC valves listed will have power

b. REC-MO-711 and REC-MO-713 will have power

c. REC-MO-713 and REC-MO-714 will have power

d. REC-MO-702, REC-MO-704, and REC-MO-714

ANSWER: RO 4

c. REC-MO-713 and REC-MO-714 will have power

Following the loss of all offsite power, 4160V bus 1F and1G are de-energized, resulting in a loss of power to MCC-S and MCC-K and the subsequent trip of all running REC pumps. After power is restored from the 'B' EDG MCC-S will be repowered since it is one of 4 MCCs on Division 2 that does not load shed. MCC-S powers MCC-Y (for valve REC-MO-714) and MCC-Rb (for valve REC-MO-713, therefore these valves power is restored.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 5	609	3	06/12/2007	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	1	Multiple Choice	BANK

Topic Area	Description
Systems	COR022102, Inter-system LOCA concern

Related Lessons	
COR0022102	REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001080I	Given a specific RPS malfunction, determine the effect on any of the following: Reactor coolant system integrity

Related References
2.1.5 10CFR55.41(b)7

Related Skills (K/A)	
212000.K3.08	Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor coolant primary system integrity (3.6/3.8)

QUESTION: RO 5

An automatic reactor Scram has occurred.

- "A" RPS scram groups 1 and 4 fail to de-energize
- All backup scrams valve fail to function.

What could result in this configuration if it is not corrected.

a. No control rods move.

b. Only part of the control rods insert.

c. The “A” Scram discharge volume will drain to “E” Sump due to the SDV drain valve being open.

d. A reactor coolant system to secondary containment flow path exists.

ANSWER: RO 5

d. A reactor coolant system to secondary containment flow path exists.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 6	9620	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Systems	Loss of RHR SDC on Fuel Pool Cooling assist

Related Lessons
COR0010602 Fuel Pool Cooling

Related Objectives
COR0010602001050B Describe the interrelationship between the FPC system and the following: Residual Heat Removal

Related References
2.4SDC 10CFR55.41(b)4

Related Skills (K/A)
205000.K3.05 Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: (CFR: 41.7 / 45.4) Fuel pool cooling assist: Plant-Specific (2.6/2.7)

QUESTION: RO 6

The plant is in day 6 of a refueling outage with RHR 'B' in SDC and crosstied with Fuel Pool Cooling (FPC) with FPC pump 1A running.

During electrical work RPS 'A' becomes de-energized.

What is the long term effect of this on the Fuel Pool?

a. No effect on the Fuel Pool.

b. Fuel Pool Cooling is partially lost due to the loss of FPC pump 1A, but RHR 'B' SDC-SFP Cooling will continue to run.

c. Fuel Pool Cooling is partially lost due to the loss of RHR 'B' SDC-SFP Cooling mode, and FPC pump 1A continues to run.

d. Fuel Pool Cooling is completely lost due to the loss of RHR 'B' SDC-SFP Cooling mode, and FPC pump 1A trips.

ANSWER: RO 6

c. Fuel Pool Cooling is partially lost due to the loss of RHR 'B' SDC-SFP Cooling mode, and FPC pump 1A continues to run.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 7	2858	00	08/30/1999	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	1	Multiple Choice	BANK

Topic Area	Description
Systems	COR0020602001090D Low Pressure Core Spray

Related Lessons	
COR0020602	CORE SPRAY

Related Objectives	
COR0020602001090D	Predict the consequences of the following items on the Core Spray System: Valve Openings

Related References
10CFR55.41(B)7

Related Skills (K/A)	
209001.K4.10	Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Testability of all operable components (2.8/2.9)

QUESTION: 7 2858 (1 point(s))

The plant is operating at near rated power with Core Spray loop A injection valves CS-MO-11 and CS-MO-12 closed due to surveillance testing. A steam leak in the drywell causes a high drywell pressure of 5 psig. Reactor pressure is decreasing due to the leak. What is the status of Core Spray loop A injection valves as reactor pressure reaches 450 psig?

a. Both valves begin opening at the same time.

b. Outboard injection valve (MO-11) opened on the high drywell pressure signal and inboard injection valve (MO-12) is now starting to open.

c. Outboard injection valve (MO-11) is now starting to open, and inboard injection valve (MO-12) will begin opening once MO-11 is full open.

d. Both valves should already be open because they started to open when the high drywell pressure signal was received.

ANSWER: 7 2858

a. Both valves begin opening at the same time.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 8	24763	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	2	Multiple Choice	NEW

Topic Area	Description
	NRC 2009 HOLD

Related Lessons
COR0023402 Alternate Shutdown

Related Objectives
COR0023402001020A Describe the interrelationship between ASD and the following: Nuclear Pressure Relief (NPR) system

Related References
10CFR55.41(b)7

Related Skills (K/A)
239002.K4.05 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) Allows for SRV operation from more than one location: Plant-Specific (3.6 3.7)

QUESTION: RO 8

Following a toxic gas event requiring the control room to be abandoned, the following conditions existed:

- The MSIVs are closed.
- All control rods are fully inserted.
- BOTH Low-Low Set valves are cycling.
- No other actions have been taken outside the control room

The CRS has directed you to take manual control of SRVs from the Alternate Shutdown (ASD) Room and maintain Reactor pressure 600-800 psig.

(NO other actions have been taken outside the control room)

The __________ SRVs can be controlled from the ASD Room.

a. 71A, 71C, and 71E

b. 71B, 71D, and 71F

c. 71A, 71B, and 71C

d. 71E, 71F, and 71G

ANSWER: RO 8

d. 71E, 71F, and 71G

EXPLANATION: Placing the Isolation switch in ISOLATE will allow operation of relief valves 71E, 71F, 71G manually from the ASD room.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 9	518	05	09/16/2008	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0021102, HPCI Response to flow signal loss

Related Lessons	
COR0021102	OPS High Pressure Coolant Injection System

Related Objectives	
COR0021102001100M	Predict the consequences of the following on the HPCI system: Inadequate system flow

Related References	
COR0021102	High Pressure Coolant Injection
10CFR55.41(b)7	

Related Skills (K/A)	
206000.K5.05	Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM: (CFR: 41.5 / 45.3) Turbine speed control: BWR-2,3,4 (3.3/3.3)

QUESTION: RO 9

A plant transient occurred which resulted in an automatic initiation of the HPCI system. One minute after HPCI reaches full flow conditions, the HPCI sensed flow signal input to the HPCI Flow Controller fails to a "0" value. No Operator action is taken.

How does the loss of the flow signal effect the operation of HPCI?

The loss of the flow signal results in . . .

a. a minimum GEMAC controller output signal which will cause the turbine RPM to lower to idle speed.

b. a maximum GEMAC controller output signal which will cause the turbine to speed up to approximately 4000 RPM.

c. the GEMAC controller transferring to MANUAL at its previous output value. Turbine speed remains the same.

d. a maximum GEMAC controller output signal which causes the governor valves to fully open. The turbine speed will increase above the overspeed setpoint.

ANSWER: RO 9

b. a maximum GEMAC controller output signal which will cause the turbine to speed up to approximately 4000 RPM.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 10	24756	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	NEW

Topic Area	Description
Systems	HOLD NRC 2009

Related Lessons
COR0030302 Sensors and Detectors (GP)

Related Objectives
COR00303020002400 Explain how the life expectancy of a neutron fission chamber is extended, and failure modes of gas-filled detectors.

Related References
10CFR55.41(b)6

Related Skills (K/A)
215003.K5.01   Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : (CFR: 41.5 / 45.3) Detector operation (2.6/2.7)

QUESTION: RO 10

The plant is in a start up on range 4 of the IRMs when IRM D fails upscale. Which of the following could be the cause of the failure?

a. Water leakage into the detector
b. Voltage lowering in the detector
c. Loss of gas pressure in the detector
d. The uranium lining in the detector has depleted

ANSWER: RO 10

a. Water leakage into the detector

EXPLANATION:

Wetting - Short circuits due to leaking of water into the detector causes the detector to fail high.
Loss of voltage - A loss of high voltage DC to the detector causes the detector to fail to a low value.
Loss of detector gas pressure - The indication is dependent on the number of primary and secondary ionizations that occur from the ionizing event. If the gas pressure decreases, there will be less gas atoms to undergo ionization, therefore less electrons collected. This means a smaller electrical current will exist and the indication will fail downscale.
Detector uranium depletion occurs slowly over the life of the detector and would result in a lower reading then actual.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	3982	02	02/23/2009	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0011702, Air loss effect on Air Compressor cooling

Related Lessons	
COR0011702	Plant Air
COR0012402	OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001040D	Describe the TEC design feature(s) and /or interlock(s) that provide for the following: Air compressor Cooling water flow control
COR0011702001070B	Given a specific Plant Air system malfunction, determine the effect on any of the following: Air Compressors
COR0012402001020H	Describe the interrelationships between the TEC system and the following: Plant Air system

Related References	
2.2.59	Plant Air System
10CFR55.41(b)7	

Related Skills (K/A)	
300000.K6.07	Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: (CFR: 41.7 / 45.7) Valves (2.5/2.6)

QUESTION: RO 11

The plant is in MODE 5 when Instrument Air isolation valve IA-1936 (air supply to the Plant Air Compressor Cooling Water Valves) is inadvertently closed.

What is the status of Air Compressor cooling?

a. Compressor "A" is cooled by TEC.
Compressors "B" AND "C" are cooled by REC.

b. Compressors "A" AND "B" are cooled by REC.
Compressor "C" is cooled by TEC.

c. Compressor "A" is cooled by REC.
Compressors "B" AND "C" are cooled by TEC.

d. ALL Air Compressors are without cooling.

ANSWER: RO 11

b. Compressors "A" AND "B" are cooled by REC.
Compressor "C" is cooled by TEC.

Explanation:

If air is lost to the REC and TEC Air Operated valves, the valves will fail such that REC will supply compressors A & B and TEC will supply compressor C.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 12	17988	00	05/05/2002	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0010102001070B, COR0020302001240F

Related Lessons	
COR0010102	AC Electrical Distribution
COR0020302	OPS CONTAINMENT

Related Objectives	
COR0010102001070B	State the electrical power supplies to the following: RPS Alternate Power
COR0020302001240F	Predict the consequences of a malfunction of the following on PCIS: RPS.

Related References	
COR0020302	Containment
COR0010102	AC Electrical Distribution
2.1.22	Recovering From A Group Isolation
4.5	Reactor Protection/Alternate Rod Insertion Systems
4.9	Primary Containment And Reactor Vessel Isolation System
10CFR55.41(b)7	

Related Skills (K/A)	
223002.K6.08	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: (CFR: 41.7 / 45.7) Reactor protection system (3.5/3.7)

QUESTION: RO 12

A plant startup is in progress with the following conditions:

- Power is 10% and steady.
- Reactor pressure is 926 psig.

The following annunciators alarm:

- C-1/G-6 BKR 1FS AUTO CLOSURE NOT PERMITTED
- C-2/A-9, STARTUP TRANSFORMER LOCKOUT
- C-2/C-10, EMERGENCY TRANSFORMER UNDERVOLTAGE
- C-4/G1 BKR 1GS AUTO CLOSURE NOT PERMITTED

Due to the change, what is the immediate plant response (if any)?

a. ½ Group 1 isolation signal

b. Group 4 and 5 isolation signals

c. No group isolation signals

d. Groups 1, 2, 3, 6, and 7 isolation signals

ANSWER: RO 12

e. Groups 1, 2, 3, 6, and 7 isolation signals

Explanation:
Following the startup transformer lockout and emergency transformer undervoltage, 4160V bus 1F and 1G are deenergized until EG-1 and EG-2 close after DG-1 and DG-2 reach rated speed and voltage; thus, MCC-L and MCC-T are deenergized for ten seconds disconnecting RPS MG sets A and B from their power supplies which results in Group 1, 2, 3, 6, and 7 isolations.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 13	24747	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Systems	COR0020802, DG Governor Failure Effect on DG Current

Related Lessons	
COR0020802	OPS DIESEL GENERATORS

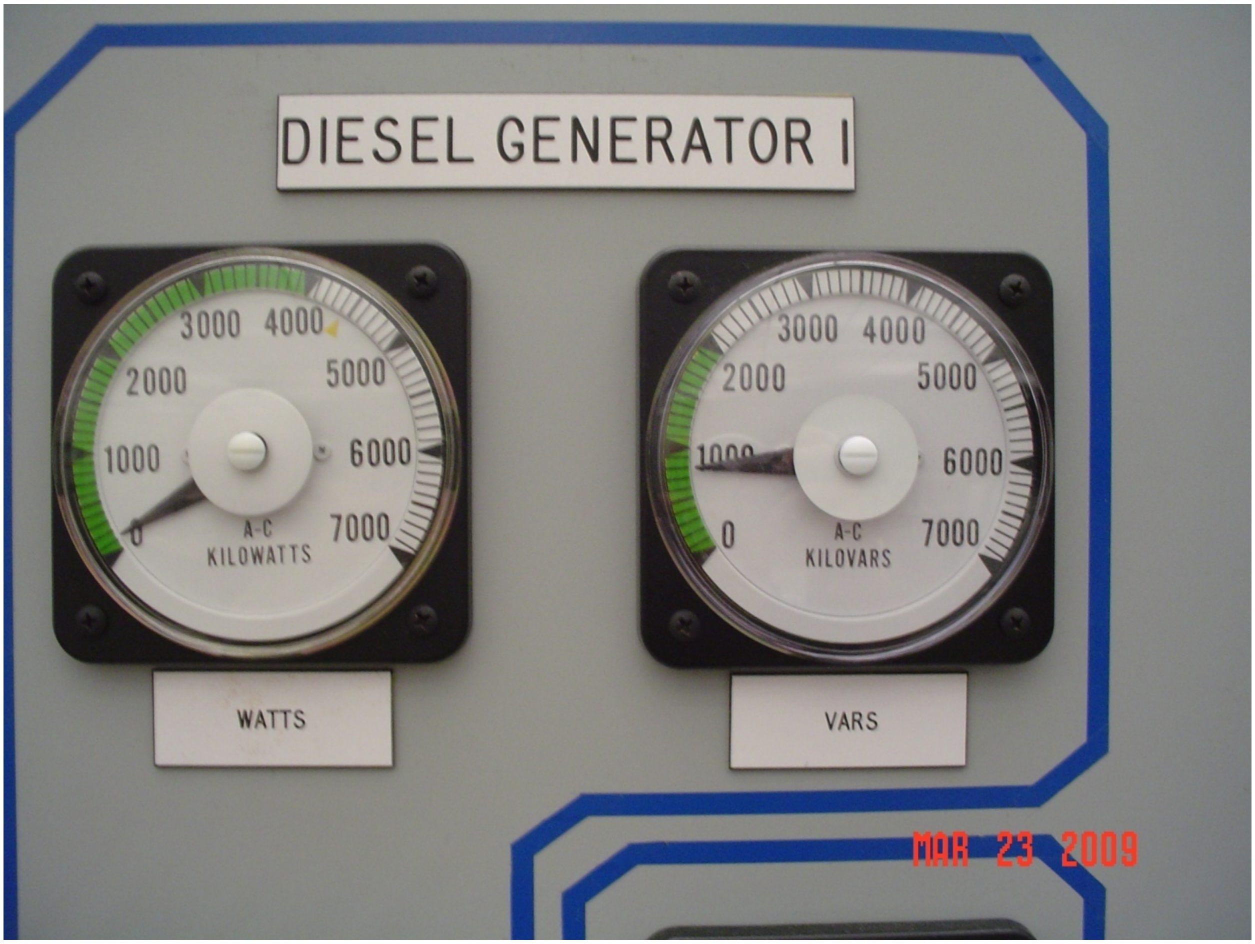
Related Objectives	
COR0020802001130A	Predict the consequences of the following items on the Diesel Generator: Parallel Operation of Diesel Generator
COR0020802001140C	Given plant conditions, determine if the following should occur: Diesel Generator trip

Related References	
2.2.20	Standby AC Power System (Diesel Generator)
10CFR55.41(b)5	

Related Skills (K/A)	
264000.A1.09	Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: (CFR: 41.5 / 45.5) Maintaining minimum load on emergency generator (to prevent reverse power) (3.0/3.1)

QUESTION: RO 13

DG-1 is paralleled to 4160V Bus 1F for surveillance testing. The following diesel generator parameters were noted:



The control room operator places and holds the DG governor to lower.

What effect does this have on DG current and what eventually causes the DG to trip?

DG current...

a. first lowers then rises. The diesel trips on overcurrent.

b. continuously increases. The diesel trips on overcurrent.

c. continuously lowers. The diesel trips on reverse power.

d. first lowers then rises. The diesel trips on reverse power.

ANSWER: RO 13

d. first lowers then rises. The diesel trips on reverse power.

With the governor in the lower position and the DG parallel to the bus, DG real load is lowering which results in a lowering current on the DG. Current continues to lower until DG load is 0. After load passes through zero (0) current now starts to rise as the DG becomes a load on the bus. Now that the DG is a load it eventually trips on reverse power.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 14	24764	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	2	Multiple Choice	NEW

Topic Area	Description
None	NRC 2009 HOLD

Related Lessons
SKL0124208 Diesel Generator

Related Objectives	
SKL012420800B0600	Comply with all related DG system limits and precautions.

Related References
10CFR55.41(b)5

Related Skills (K/A)	
262001.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5) Load currents (2.7/2.9)

QUESTION: RO 14

The plant was operating near rated power when a load reject and Loss of Offsite Power occurred.

ATWS conditions exist with Reactor power approximately 25%
EDG A is loaded to 4000 KW
EDG B can not be started
The CRS has ordered Maximize Suppression Pool Cooling

Based on the expected increase load on EDG A for SP Cooling, how long can this condition exist?

a. 2 hours based on USAR limits on EDG loading

b. 2 hours based on 5.3GRID limits on EDG loading

c. 4 hours based on USAR limits on EDG loading

d. 4 hours based on 5.3EMPWR limits on EDG loading

ANSWER: RO 14

a. 2 hours based on USAR limits on EDG loading

EXPLANATION OF ANSWER:

The allowable time limit for overload condition is defined in 5.3EMPWR and USAR. This states the EDG can be run in the overload condition for 2 hours out of any 24 hour period.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 15	20669	00	04/26/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080607, Steam Cooling Conditions Requiring ED

Related Lessons	
INT0080607	OPS EOP Flowchart 2A - Emergency RPV Depressurization & Steam Cooling

Related Objectives	
INT00806070010800	Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, state the reasons for the actions contained in the steps.
INT0080607001050B	State the basis for Emergency Depressurization actions as they apply to: Pressure Control Systems

Related References
10CFR55.41(b)10

Related Skills (K/A)	
218000.A2.03	Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Loss of air supply to ADS valves: Plant-Specific (3.4/3.6)

QUESTION: RO 15

The plant was operating at power when multiple failures resulted in no injection systems, injection subsystems or alternate injection subsystems being available. The crew entered EOP-2A, Steam Cooling and stabilized reactor pressure using RCIC in the pressure control mode. (RCIC is available for pressure control ONLY).

The following reactor conditions exist:

- Reactor pressure is 900 psig and rising
- Reactor water level is -200" (FZ indicated) and slowly lowering
- IA and Nitrogen to the drywell has just been lost and cannot be recovered.
- All SRVs are closed.
- RPS ‘A’ and ‘B’ have been lost and can not be restored.

The RCIC turbine tripped and cannot be restarted due to trip throttle valve failure.

What action is required and why?

a. Emergency depressurize the reactor while pneumatics are available to SRV valves.

b. Place RWCU in the blowdown mode to control reactor pressure as it remains the only pressure control system available.

c. Emergency Depressurize the reactor because level is less than the minimum zero injection reactor water level (MZIRWL).

d. Control reactor pressure by allowing the SRVs to cycle on mechanical setpoints to prevent the depletion of ADS valve accumulator pressure.

ANSWER: RO 15

a. Emergency depressurize the reactor while pneumatics are available to SRV valves.

The loss of RCIC leaves only the SRVs for pressure control. EOP-2A would be entered in Steam Cooling Mode. Since SRVs are the only pressure control remaining and a continuous supply of IA/N2 is not available, an ED is required. (Override in RC/P-16.) The mechanical setpoint for the SRVs are well above the current reactor pressure. This could invalidate the MZIRWL calculations because the change in core inlet subcooling could potentially reduce the steam flow past the upper portion of the core.

PROVIDE EOP 1A and 2A with Entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 16	7746	02	02/24/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0012402, TEC 1A Pump Disch clog - What is affect on system?

Related Lessons	
COR0012402	OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001040A	Describe the TEC design feature(s) and /or interlock(s) that provide for the following: TEC automatic temperature control.
COR0012402001070E	Predict the consequences the following would have on the TEC system: Loss of TEC pumps

Related References
10CFR55.41(b)5

Related Skills (K/A)	
400000.A2.03	Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) High/low CCW temperature (2.9/3.0)

QUESTION: RO 16

It is May with the plant operating at rated power with 1A and 1C TEC pumps in operation. The 1A TEC Pump discharge becomes clogged, passing little to no flow.

What is the final effect on the TEC system and what actions are required per the procedure?

a. The TEC pump 1B auto starts due to system low pressure; no further actions required.

b. TEC system temperature lowers due to a lower flow through the TEC heat exchanger, the operator should throttle open on the TEC Heat Exchanger bypass.

c. The TEC system TCV on the outlet of the TEC heat exchanger opens to compensate for the lower TEC flow; the operator should start 1B TEC pump.

d. The automatic TCVs throughout the TEC system open to compensate for the resultant higher TEC temperature; the operator should start 1B TEC pump.

ANSWER: RO 16

d. The automatic TCVs throughout the TEC system open to compensate for the resultant higher TEC temperature; the operator should start 1B TEC pump.

The operator must manually start the 1B TEC pump due to the TCV valves throughout the system opening as temperatures on cooled components go up.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 17	24258	00	06/26/2008	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	5	Multiple Choice	BANK

Topic Area	Description
Systems	COR0023202001040A, COR0023202001060K, COR0020702001080O

Related Lessons	
COR0023202	OPS REACTOR VESSEL LEVEL CONTROL
COR0020702	OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0023202001040A	State the electrical power supplies to the following: RVLC Components
COR0023202001060I	Predict the consequences of the following on the RVLC system: Loss of RPV water level input
COR0020702001080O	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor feedwater system

Related References
10CFR55.41(b)7

Related Skills (K/A)	
259002.A3.04	A3. Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: (CFR: 41.7 / 45.7) Changes in reactor feedwater flow (3.2/3.2)

QUESTION: RO 17

The plant is operating at 92% power when MCBP ‘A’ trips.

What is the expected plant response?

a. RR pumps runback toward 45% and RVLCS continues to control level in 3 element control.

b. RR pumps will not runback toward 45% and RVLCS will continue to control level in 3 element control.

c. RR pumps runback toward 45% and RVLCS will transfer to single element control.

d. RR pumps will not runback toward 45% and RVLCS will transfer to single element control.

ANSWER: RO 17

a. RR pumps will runback toward 45% and RVLCS continues to control level in 3 element control.

EXPLANATION OF ANSWER: At 92 % power, steam flow is approximately 8.7 mil lbm/hr, which is > 8.25mil lbm/hr so RR pumps will runback . RVLCS remains in 3 element for this condition, these conditions will not result in auto transfer to single element control.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 18	1102	01	09/06/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	5	Multiple Choice	BANK

Topic Area	Description
Systems	COR0010102, NBPP inverter voltage failure alarms, what is the final source of power to the NBPP?

Related Lessons	
COR0010102	AC Electrical Distribution

Related Objectives	
COR0010102001060C	Describe the interrelationship between the AC Electrical Distribution System and the following: No Break Power Supply

Related References	
2.2.22	Vital Instrument Power System
2.3_C-4	Panel C - Annunciator C-4
10CFR55.41(b)7	

Related Skills (K/A)	
262002.A3.01	Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: (CFR: 41.7 / 45.7) Transfer from preferred to alternate source (2.8/3.1)

QUESTION: RO 18
Annunciator C-4/E-7 NO BREAK SYSTEM INVERTER 1A VOLTAGE FAILURE alarm has been received on the No Break Power Panel (NBPP) Inverter.

What is the source of power to the NBPP for these conditions?

Power to the No Break Power Panel will come from…

a. MCC-R.

b. MCC-RA.

c. 250 VDC Bus 1A.

d. 250 VDC 1B.

ANSWER: RO 18

a. AC power due to static switch operation.

EXPLANATION OF ANSWER: The inverter failure alarm indicates that the power into or out of the inverter is failed causing the NBPP to transfer to MCC-R. b. 250 VDC is the normal power supply to NBPP. c. 125 VDC is not used by the NBPP d. the static inverter is the normal source of power to NBPP

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 19	NEW	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	2	Multiple Choice	NEW

Topic Area	Description
Systems	COR0022902001130A, COR0022902001130B Standby Liquid Control System

Related Lessons	
COR0022902	OPS STANDBY LIQUID CONTROL

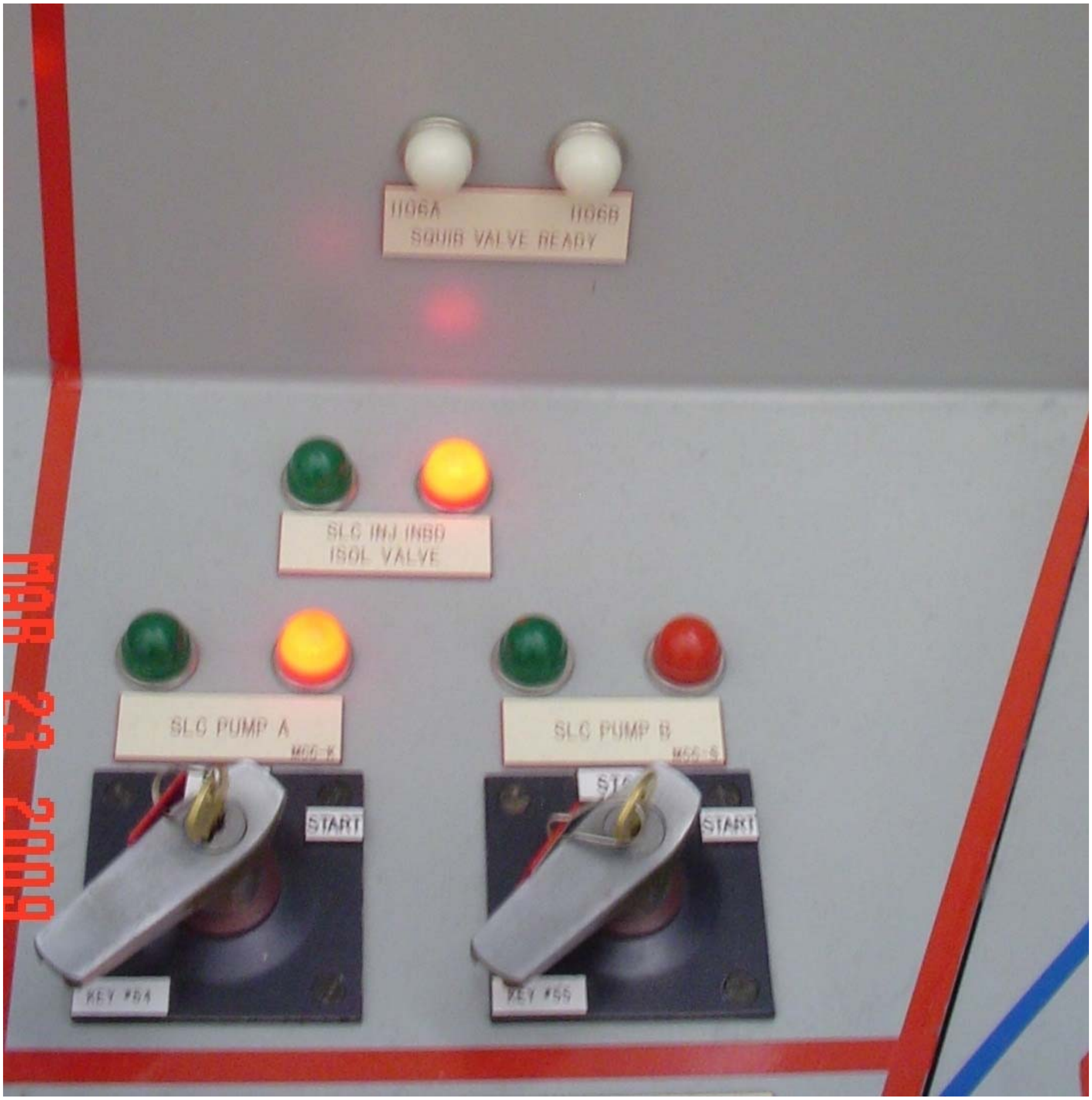
Related Objectives	
COR0022902001130A	State the electrical power supply to the following SLC components: Pump motors.
COR0022902001130B	State the electrical power supply to the following SLC components: Squib valves

Related References
10CFR55.41(b)6

Related Skills (K/A)
211000. A4.03 - Ability to manually operate and/or monitor in the control room: Explosive valves firing circuit status

QUESTION: RO 19

During ATWS conditions the following indications are noted on 9-5.



What is the status of SLC injection?

a. SLC pumps are not injecting

b. SLC Pump 1A is injecting through Squib Valve 14A only

c. SLC Pump 1A is injecting through Squib Valve 14B only

d. SLC Pump 1A is injecting through Squib Valve 14A and Squib Valve 14B

ANSWER: RO 19

d. SLC Pump 1A is injecting through Squib Valve 14A and Squib Valve 14B

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 20	23208	00	07/30/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	COR0023002, Monitor and operate SRM backpanel switches. (2006 BIENNIAL EXAM)

Related Lessons	
COR0023002	SOURCE RANGE MONITOR

Related Objectives	
COR0023002001060A	Describe the SRM system design features and/or interlocks that provide for the following: Rod withdrawal blocks

Related References
10CFR55.41(b)6

Related Skills (K/A)	
215004.A4.07	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Verification of proper functioning/ operability (3.4/3.6)

QUESTION: RO 20

The following conditions exist:

- A plant startup is in progress.
- Reactor is critical with power on Range 2 of the IRMs.
- SRM A indicates 2.5E4 CPS.
- SRM A detector is fully inserted.

Testing needs to be performed on SRM A. This testing requires that the SRM A Function Switch be placed to STANDBY position.

What additional condition or action would prevent the generation of a Rod Block while the switch is in STANDBY?

a. Partially withdraw SRM A detector.

b. Raise reactor power to Range 3 of the IRMs.

c. Ensure the Reset Switch on Panel 9-12 is held in the left position while the Function switch is in STANDBY.

d. SRM A INOP INHIBIT pushbutton on Panel 9-12 is continually depressed while the Function switch is in STANDBY.

ANSWER: RO 20

d. SRM A INOP INHIBIT pushbutton on Panel 9-12 is continually depressed while the Function switch is in STANDBY.

To bypass the Function switch INOP and allow for moving the Function switch to a position other than "OPERATE", without initiating an alarm or rod block from an SRM "INOP" the INOP INHIBIT pushbutton is depressed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 21	9635	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	NEW

Topic Area	Description
Systems	EOP Entry

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE
INT0320135	CNS Abnormal Procedures (RO) - Condensate/Feedwater

Related Objectives	
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.
INT0320135G0G0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.41(b)10

Related Skills (K/A)
206000.2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (4.5/4.6)

QUESTION: RO 21
The plant is operating at near rated power when the following occured:

- 'A' and 'B' RFP tripped
- Reactor water level has lowered and HPCI has started
- HPCI injection is restoring Reactor water level

Along with 2.1.5 Reactor Scram these are entry conditions for what additional procedures?

A. 2.4MC-RF only
B. 2.4MC-RF, and 2.4RXLVL only
C. 2.4MC-RF and EOP 1A only
D. 2.4MC-RF, 2.4RXLVL, and EOP 1A

ANSWER: RO 21

C. 2.4MC-RF and EOP 1A only

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 22	5464	00	11/17/1999	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	Standby Gas Treatment

Related Lessons	
COR0022802	OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001050A	Describe the interrelationships between SGT and the following: Reactor Building Ventilation System

Related References	
2.2.73	Standby Gas Treatment System
10CFR55.41(b)7	

Related Skills (K/A)	
261000.2.4.8	Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10/43.5/45.13) (3.8\4.5)

QUESTION: RO 22

Given the following conditions:

- Standby Gas Treatment (SGT) System initiates
- Reactor Building Ventilation (RBV) isolates
- High Pressure Coolant Injection (HPCI) auto initiates
- Reactor Core Isolation Cooling (RCIC) does NOT receive an automatic initiation signal

Which of the following statements identifies the cause of the SGT System initiation?

a. Low RPV water level

b. High Drywell pressure

c. Low Reactor Building differential pressure

d. High Reactor Building Ventilation radiation

ANSWER: RO 22

b. High Drywell pressure

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 23	2256	01	08/12/2004	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0010102, Degraded Voltage effect on 1F/1G and reactor scram or not

Related Lessons	
COR0010102	AC Electrical Distribution

Related Objectives	
COR0010102001080B	Predict the consequences of the following on plant operation: 4160V Critical bus undervoltage
COR0010102001150A	Briefly describe the following concepts as they apply to AC Electrical Distribution System: Load shedding

Related References	
5.3GRID	Degraded Grid Voltage
10CFR55.41(b)7	

Related Skills (K/A)	
262001.K6.02	Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: (CFR: 41.7 / 45.7) Off-site power (3.6/3.9)

QUESTION: RO 23

The plant is operating at rated power when 345 KV, 161 KV and 69 KV voltages simultaneously lower such that the Normal Transformer, Startup Transformer and Emergency Transformer secondary voltages drop to 3700 VAC.

What plant conditions would exist after 1 minute if this voltage reduction condition persists? (No operator actions occur during the 1 minute).

4160 1F and 1G are supplied by the . . .

a. Diesel Generators and the reactor is scrammed.

b. Diesel Generators and the reactor is NOT scrammed.

c. Emergency Transformer and the reactor is scrammed.

d. Emergency Transformer and the reactor is NOT scrammed.

ANSWER: RO 23

a. Diesel Generators and the reactor is scrammed.

Explanation:

With the NSST supplying 1A/1F & 1B/1G and bus voltage lowering to 3700 volts, breakers 1FA and 1GB will trip 12.5 seconds after voltage has lowered below 3880 VAC. The "loss of voltage" signal will start the Diesel generators and apply a close permissive to 1FS & 1GS. As the Emergency Transformer voltage is also degraded, 1FS/1GS will not automatically close and the EDGs will close onto the bus when they have reached rated voltage and speed (at least 10 seconds after the bus was de-energized).

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 24	14528	02	04/22/2007	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	5	Multiple Choice	BANK

Topic Area	Description
Systems	COR0021602001060A, ADS timer timed out w/o LP pump.

Related Lessons	
COR0021602	OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001060A	Briefly describe the following concepts as they apply to NPR: ADS logic operation

Related References	
10CFR55.41	Written examinations: Operators
791E253	Automatic Blowdown System
791E261	RHR Elementary Diagram
791E265	Core Spray Elementary Diagram
10CFR55.41(b)7	

Related Skills (K/A)	
218000.2.1.23	Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10/43.5/45.2/45.6) (4.3/4.4)

QUESTION: RO 24

The plant was operating at 50% power following the loss of AA-2 when a loss of coolant accident occurred concurrent with a loss of offsite power.

- Neither DG automatically started and crew continued attempts to manually start the diesels are unsuccessful.
- At 1400 Reactor water level lowered to -113" (WR).
- At 1405 the crew restored power to 4160V 1F from the emergency transformer.
- The ADS Inhibit Switches are in AUTO.

The ONLY low pressure pumps the crew were able to start were CS 1A and RHR 1B.

How does ADS system respond to the start of these pumps?

ADS actuates...

a. immediately after the start of CS pump 1A.

b. 109 seconds after the start of CS pump 1A.

c. immediately after the start of RHR pump 1B.

d. 109 seconds after the start of RHR pump 1B.

ANSWER: RO 24

c. immediately after the start of RHR pump 1B.

Explanation:

When reactor water level dropped below -113" the ADS timers actuated, but since no CS or RHR pump was running ADS will not actuate. The loss of AA-2 resulted in the loss of ADS "A" logic (so CS pump 1A will not fire the ADS logic if it could be started) but the "B" logic remains powered and capable of initiating a blowdown. The start of RHR pump 1B would result in the immediate actuation of ADS because the ADS timers have timed out.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 25	4572	1	06/18/2002	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	2	Multiple Choice	BANK

Topic Area	Description
Administrative	General Operating Procedure 2.1.1, Startup Procedure

Related Lessons
INT0320104 CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives
INT032010400A0300 Describe the general sequence of events performed during a Reactor startup and RPV heatup.

Related References
10CFR55.41(b)10

Related Skills (K/A)
215003.A4.07 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Verification of proper functioning/ operability (3.6/3.6)

QUESTION: RO 25

Per General Operating Procedure 2.1.1, "Startup Procedure", how is Source Range Monitor (SRM) to Intermediate Range Monitor (IRM) overlap verified?

By observing all operable IRM channels . . .

a. indicate on-scale, prior to operable SRMs reaching 1E6 cps, with SRM detectors fully inserted.

b. indicate on-scale prior to operable SRMs reaching 1E6 cps and prior to SRM detectors being fully withdrawn.

c. have cleared their downscale alarms, with all operable SRMs reading between 1E3 and 1E5 cps, with SRM detectors fully retracted.

d. have cleared their downscale alarms with all operable SRMs reading between 1E3 and 1E5 cps and prior to SRM detectors being fully withdrawn.

ANSWER: RO 25

a. indicate on-scale, prior to operable SRMs reaching 1E6 cps, with SRM detectors fully inserted.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 26	3832	01	08/19/2004	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	SKL0124212A0A0200 Intermediate Range Monitor System

Related Lessons
SKL0124212 INTERMEDIATE RANGE MONITOR SYSTEM COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives
SKL012421200A0200 Explain the IRM system limitations and precautions as stated in the IOP 4.1.2, IRM System. COR0021202001090A Given plant conditions, determine if the following IRM actions should occur: Rod Block. COR0021202001090B Given plant conditions, determine if the following IRM actions should occur: Reactor Scram. 10CFR55.41(b)2

Related References
4.1.2 Intermediate Range Monitoring System

Related Skills (K/A)
212000.K4.11 Knowledge of REACTOR PROTECTION SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Operation with shorting links removed: Plant-Specific (3.3/3.5)

QUESTION: RO 26

The following condition exists:

- A plant startup is in progress.
- The Reactor Mode Switch is in STARTUP.
- "C" IRM fails upscale while on range 3 AND results in a full scram.
- All control rods fully insert on the scram.

What condition resulted in this response?

The full scram occurred due to . . .

a. the IRM being on the 0-40 scale.

b. the associated APRM being downscale.

c. ALL four shorting links being closed.

d. ALL four shorting link switches being open.

ANSWER: RO 26

d. ALL four shorting link switches being open.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 27	19341	01	07/08/2003	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR00222020010200, Recirc Flow Mismatch

Related Lessons	
COR0022202	REACTOR RECIRCULATION

Related Objectives	
COR00222020010200	Given conditions and/or parameters associated with the Reactor Recirculation system or the Recirculation Flow Control system, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operation are met.

Related References
10CFR55.41(b)6

Related Skills (K/A)	
202002.K1.01	Knowledge of the physical connections and/or cause/effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9/45.7 to 45.8) Recirculation system (3.5/3.6)

QUESTION: RO 27

The plant was at power when the B Recirc Pump Controller output was slowly rising. The RO locked the scoop tube. Reactor power is currently 100%. Loop A and Loop B Jet Pump Flow instruments indicate:

- NBI-FI-92A = 32 x 10^6 lbs/hr
- NBI-FI-92B = 37 x 10^6 lbs/hr

What operational concern due you have with current plant conditions and what action can you direct to address this concern?

a. continued operation will be outside the bounds of the LOCA accident analysis; the speed of A Recirc Pump should be raised.

b. continued operation will be outside the bounds of the LOCA accident analysis; the speed of B Recirc Pump should be lowered locally.

c. Continued operation may caused high vibration and damage to the jet pump risers; the speed of A Recirc Pump should be raised.

d. Continued operation may cause high vibration and damage to the jet pump risers; the speed of B Recirc Pump should be lowered locally.

ANSWER: RO 27

b. continued operation will be outside the bounds of the LOCA accident analysis; the speed of B Recirc Pump should be lowered locally.

Vibration and damage to the jet pumps only occurs with very large mismatchs where reverse flow occurs in the lower speed loop. B Recirc Pump speed should be lowered locally.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 28	19317	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	4	Multiple Choice	NEW

Topic Area	Description
Systems	COR0022302,RESIDUAL HEAT REMOVAL; AC Power loss

Related Lessons	
COR0022302	RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030A	Describe RHR System design feature(s) and/or interlocks which provide for the following: Automatic initiation/injection
COR0022302001030B	Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of piping over-pressurization
COR0022302001030C	Describe RHR System design feature(s) and/or interlocks which provide for the following: Pump minimum flow protection
COR0022302001030D	Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of water hammer
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Related References
10CFR55.41(b)7

Related Skills (K/A)	
230000.K2.02	K2. Knowledge of electrical power supplies to the following: (CFR: 41.7) Pumps (2.8*/2.9*)

QUESTION: RO 28

The following plant conditions exist:

- A LOCA signal is present.
- RHR Loop A is aligned for Torus Sprays.
- RHR Loop B is aligned for Drywell Sprays.
- The Startup Station Service Transformer lockout actuates (Alarm C-2/A-9)

Which of the following describes the impact on the RHR System?

RHR pumps trip and ...

a. automatically restart when power is restored; spray valves close.

b. automatically restart when power is restored; spray valves remain open.

c. must be manually restarted when power is restored; spray valves close.

d. must be manually restarted when power is restored; spray valves remain open.

ANSWER: RO 28

b. is correct. Since the LOCA signal is present the pumps will automatically start when power restored by the diesel generator. The spray valves will lose power but they will fail as is on the lost of power.

a. is incorrect. The spray valves do not repostion on a loss of AC power and they have not lost their initiation signals to cause them to close. RHR initiation logic is 125 VDC and is not loss.

c.and d. are incorrect. The pumps will automatically restart upon restoration of AC power.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	2102	00	08/12/1999	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	COR0020502, Control Rod Drive Mechanisms

Related Lessons	
COR0020502	CONTROL ROD DRIVE MECHANISM

Related Objectives	
COR0020502001090A	Predict the consequences a malfunction of the following would have on the CRDMs: Loss of CRDH Pumps
COR0020502001090B	Predict the consequences a malfunction of the following would have on the CRDMs: Loss of Reactor Pressure

Related References
10CFR55.41(b)6

Related Skills (K/A)	
201001.K3.03	Knowledge of the effect that a loss or malfunction of the CONTROL ROD DRIVE HYDRAULIC SYSTEM will have on following: (CFR: 41.7 / 45.4) K3.03 Control rod drive mechanisms (3.1/3.2)

QUESTION: RO 29

During a startup with reactor pressure at 250 psig, "A" CRD pump trips off and "B" CRD pump breaker will not close in when tried. What is the immediate major concern?

a. Excessive scram times due to elevated CRDM temperatures.

b. Insufficient pressure to insert the control rods in the event of a scram.

c. Seal damage to the reactor recirculation pumps due to a loss of mini-purge.

d. Seal damage to the reactor water cleanup pumps due to loss of mini-purge.

ANSWER: RO 29

b. Insufficient pressure to insert the control rods in the event of a scram.

Answer b is correct. The loss CRD pumps will eventually lead to a loss of accumulator pressure. At this reduced reactor presssure insufficient pressure is available to complete a scram. Answer a is incorrect even though excessive CRDM temperatures due lead to excessive scram times that is not an immediate major concern compared to an inability to promptly insert the control rods on a scram. In addition, at this point in the startup the temperature in the vessel is low enough that it would take a much longer time to reach temperatures that could lead to longer scram times. Answer c and d are incorrect because even though the loss of mini-purge is a concern with regards to seal life, this concern is not a major concern when compared to the loss of the capability to scram.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 30	22923	00	03/19/2007	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	DEH response with Digital Computer BIAS placed in BYPASS

Related Lessons	
COR0020902	Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040L	Describe how the DEH control system operates to control the following: Bypass valve position
COR0020902001040M	Describe how the DEH control system operates to control the following: Turbine governor valve position
COR0020902001040Q	Describe how the DEH control system operates to control the following: Load limit set

Related References	
2.2.77	Turbine Generator
10CFR55.41(b)7	

Related Skills (K/A)	
241000.K4.19	Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Steam bypass valve control (3.6/3.7)

QUESTION: RO 30

With the plant at 50% power and 50% steam flow, the Valve Position Limit setpoint is lowered to less than the pressure control signal.

Which of the following describes the system and/or plant response (if any) as the Valve Position Limit setpoint exceeds any bias in the system?

a. Neither Main Turbine Governor Valves nor Main Turbine Bypass Valves would change position.

b. Main Turbine Governor Valves would start to close and Main Turbine Bypass Valves would start to open to maintain Equalizing Header pressure at current setpoint.

c. Main Turbine Governor Valves would start to close and Main Turbine Bypass Valves would remain closed. The Reactor will scram on Neutron High Flux/Reactor High Pressure.

d. Main Turbine Governor Valves would start to open and Main Turbine Bypass Valves would start to open. Equalizing Header pressure would lower. The Main Steam Isolation Valves will close on PCIS Group 1 isolation. The Reactor will scram on MSIV Not Full Open logic signal.

ANSWER:

b. Main Turbine Governor Valves would start to close and Main Turbine Bypass Valves would start to open to maintain Equalizing Header pressure at current setpoint.

Justification:

Valve Position Limit applied to Main Turbine Governor Valves. As the limit causes the governor valves to close, the Main Turbine Bypass Valves, unaffected by this limit, will open to maintain Equalizing Header pressure (until the Valve Position Limit changes the position of the Main Turbine Governor Valves more than the Main Turbine Bypass Valves can compensate (i.e., 25%).

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 31	23293	00	05/22/2002	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0021502, NUCLEAR BOILER INSTRUMENTATION (2002 NRC Exam) (2006 BIENNIAL EXAM)

Related Lessons	
COR0021502	OPS NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001040K	Briefly describe the following concepts as they apply to NBI: Effects on level indication due to rapid changes in void fraction

Related References
10CFR55.41(b) 10

Related Skills (K/A)	
216000.K5.12	Knowledge of the operational implications of the following concepts as they apply to NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.5 / 45.3) Effects on level indication due to rapid changes in void fraction (3.2/3.3)

QUESTION: RO 31

The plant is operating at 50% power when a DEH malfunction causes the Main Turbine Bypass valves to rapidly open.

Initially how is flow resistance in the core region affected?

Initially how is indicated wide range reactor water level affected?

a. increase, increase

b. increase, decrease

c. decrease, increase

d. decrease, decrease

ANSWER: RO 31

a. increase, increase

The flow restriction rises due to back pressure from the Steam flow through thte moisture separator and increased boiling in the core region. The mass above the variable leg increases and indicated level rises.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 32	10408	01	01/04/2008	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Updated Safety Analysis Report	INT0060118, Accident Analysis

Related Lessons
INT0060118 Accident Analysis INT0060115 ACCIDENT ANALYSIS

Related Objectives	
INT00601180010700	Given a list of conditions, select the effect those conditions could have on the magnitude of suppression pool heat up and primary containment pressure following a LOCA (Loss of Coolant Accident) in the primary containment.
INT00601180011700	Given a specific accident, describe the long term actions required for core cooling and plant stabilization.
INT00601150010700	Given a condition or conditions, select the effect those conditions could have on the magnitude of suppression pool heatup and containment pressure following a LOCA in containment.
INT00601150010200	Given an accident and a set of conditions, select those conditions that would tend to make the consequences of the given accident more severe.

Related References
10CFR55.41(b)7

Related Skills (K/A)	
223001.K6.14	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: (CFR: 41.7 / 45.7) RHR/LPCI (3.6/3.8)

QUESTION: RO 32

A large break LOCA has occurred. In addition to Core Spray pumps, only the following equipment can be made to operate:

- 1 RHR pump
- 1 RHR Booster pump
- 1 Service Water pump
- 1 RHR heat exchanger
- Containment Spray

What is the LONG TERM effect (if any) on peak containment pressure and peak suppression pool water temperature *as compared to* the effect if ALL RHR and Service Water equipment were available?

(NOTE: "Long term" is defined for parameters as greater than 10 minutes after the event.)

With only one train of RHR, . . .

a. Only long term peak containment pressure is higher.

b. Only long term peak suppression pool water temperature is higher.

c. Both long term peak containment pressure and long term peak suppression pool water temperature are higher.

d. Neither long term peak containment pressure nor long term peak suppression pool water temperature are higher.

ANSWER: RO 32

c. Both long term peak containment pressure and long term peak suppression pool water temperature are higher.

REFERENCE: USAR, Volume 5, Chapter XIV

Distractors:

a. Suppression pool water temperature will also be higher.
b. Containment temperature will also be higher.
d. Both parameters will be higher.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 33	11388	00	01/15/2001	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022602, Effect of reed switch failure at 7% power

Related Lessons	
COR0022002	OPS REACTOR MANUAL CONTROL SYSTEM
COR0022602	OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022002001050C	Predict the consequences the following would have on the RMCS and/or RPIS: RPIS failure
COR0022002001150A	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod withdrawal block
COR0022602001140A	Given plant conditions, determine if the following RWM actions should occur: Insert Rod block.
COR0022602001140B	Given plant conditions, determine if the following RWM actions should occur: Withdrawal Rod block.

Related References
10CFR55.41(b)6

Related Skills (K/A)	
214000.K1.01	Knowledge of the physical connections and/or cause- effect relationships between ROD POSITION INFORMATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) RWM: Plant-Specific (3.0/3.2)

QUESTION: RO 33

The plant is starting up at 7% power when the reed switch at notch 48 for a non-selected, fully withdrawn control rod (rod 22-23) fails open.

What effect (if any) does this failure have on the ability to move control rods?

RWM will . . .

a. operates normally unless rod 22-23 is selected. If rod 22-23 is selected, both insertion and withdrawal are inhibited.

b. inhibits only control rod withdrawal of all control rods. Control rods CANNOT be withdrawn, even if RWM is bypassed.

c. inhibits only control rod insertion of all control rod. Control rods can be inserted ONLY if RWM is bypassed.

d. inhibits both insertion AND withdrawal of all control rods. Control rods CAN be moved if RWM is bypassed.

ANSWER: RO 33

d. inhibits both insertion AND withdrawal of all control rods. Control rods CAN be moved if RWM is bypassed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 34	23097	00	08/01/2006	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	5	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080617, Determine EOP-5A actiopns when indications of fuel failure are present. (2006 BIENNIAL EXAM)

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.

Related References	
(B)(10)	Administrative, normal, abnormal, and emergency operating procedure for the facility.

Related Skills (K/A)	
268000.A2.01	Ability to (a) predict the impacts of the following on the RADWASTE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.01 System rupture (2.9/3.5)

QUESTION: RO 34

The plant is operating at power when an accident occurred. The following conditions exist:

- Reactor coolant sample is 5.4 µCi/ml, Dose Equivelent I-131 last sample was 1.25E-4 µCi/ml.
- The reactor is scrammed and the MSIVs are closed based on 5.2Fuel.
- There is an unisolable steam leak in the steam tunnel.

In addition:

- There is a leak on the torus side of RHR-MO-13 and it cannot be isolated.
- S-1/A-3, REACTOR BLDG C SUMP HI-HI LEVEL is alarming.
- SW Quad water level is 4 ft and slowly rising.

What actions are required for the C sump pumps?

a. Immediately open the breakers for sump pumps 1C-1 and 1C-2.

b. Operate sump pumps 1C-1 and 1C-2 only as needed to maintain level less than 9.5 ft.

c. Operate sump pumps 1C-1 and 1C-2 only as needed to clear the sump C HI-HI Level Alarm.

d. Open the breakers for sump pumps 1C-1 and 1C-2 when SW quad water level approaches 9 ft.

ANSWER: RO 34

b. Operate sump pumps 1C-1 and 1C-2 only as needed to maintain level less than 9.5 ft.

Explanation:

The high coolant activity is an entry condition for 5.2FUEL and the high sump level provides an entry into EOP-5A. 5.2Fuel will direct the operator to attachment 2 which instructs the operator that the sump pumps are only operated as required to maintain level less than max safe operating level (9.5 ft). This is to minimize rad transport from secondary containment sumps to Radwaste.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 35	19315	02	05/04/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	6	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022402, RBM Response to Reference APRM failure

Related Lessons	
COR0022402	OPS ROD BLOCK MONITOR

Related Objectives	
COR0022402001130B	Given plant conditions, determine if any of the following should occur: RBM bypass.

Related References	
4.1.5	Rod Block Monitor System
10CFR55.41(b)6	

Related Skills (K/A)	
215002.A2.03	Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, or control, or...: (CFR: 41.5 / 45.6) Loss of associated reference APRM channel: BWR-3,4,5 (3.1/3.3)

QUESTION: RO 35

Given the following:

- The reactor is operating at 80% power.
- APRM Channel E is bypassed.
- Rod 22-31 is selected and is withdrawn a notch.
- RBM A and B levels are 106/125 of scale.
- APRM Channel B fails downscale.

NOTE: Use Tech Spec values when assessing this question.

What is the impact on the Rod Block Monitor system including the action required to allow continued rod withdrawal?

a. RBM A generates a rod block; bypass RBM A on Panel 9-5.

b. RBM A is automatically bypassed; bypass APRM Channel B on Panel 9-5.

c. RBM B generates a rod block; bypass RBM B on Panel 9-5.

d. RBM B is automatically bypassed; bypass APRM Channel B on Panel 9-5.

ANSWER: RO 35

d. RBM B is automatically bypassed; bypass APRM Channel B on Panel 9-5.

RBM B is automatically bypassed since its reference APRM (APRM B) is reading less than 27.5%. Bypassing APRM B due to its downscale failure will align APRM D as the reference APRM to RBM B and, more importantly, will clear the APRM downscale rod block signal and allow continued rod withdrawal.

a.and b. are incorrect. RBM A is not impacted by failure of APRM B. RBM A is currently using APRM C as its reference signal since APRM E is bypassed.

c. is incorrect. RBM B will not generate a rod block since it is bypassed automatically when its reference APRM fails to less than 27.5%. The RBM is inop but will not initiate the INOP trip; bypassing RBM B is not appropriate since it is not the failed component.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 36	NEW	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	NEW

Topic Area	Description
Systems	COR0011802, Rx Bldg monitor Grp 6 Failure

Related Lessons	
COR0011802	OPS Radiation Monitoring

Related Objectives	
COR0011802001100D	Given a control manipulation, predict and explain the changes to the following Radiation Monitoring systems: Reactor Building Vent Exhaust Plenum radiation monitoring system

Related References	
2.1.22	Recovering From A Group Isolation
10CFR55.41(b)9	

Related Skills (K/A)	
290001.A4.09	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System status lights and alarms: Plant-Specific (3.2/3.2)

QUESTION: RO 36

With the plant at full power, the following Reactor Building vent exhaust plenum radiation monitor readings exist in the control room:

- RMP-RM-452A: 14 mrem/hr
- RMP-RM-452B: 7 mrem/hr
- RMP-RM-452C: 8 mrem/hr
- RMP-RM-452D: 11 mrem/hr

Which Alarms should be displayed by RONAN? (Base answer on actual alarm setpoints.)

a. 9-4-1/E-5 (1777) RX BLDG VENT MONITOR DIV I HIGH RAD ONLY.

b. 9-4-1/E-5 (1777) RX BLDG VENT MONITOR DIV I HIGH RAD <u>AND</u>
9-4-1/E-5 (1778) RX BLDG VENT MONITOR DIV II HIGH RAD ONLY.

c. 9-4-1/E-5 (1777) RX BLDG VENT MONITOR DIV I HIGH RAD <u>AND</u>
9-4-1/E-4 (1763) RX BLDG VENT MONITOR A HI-HI RAD ONLY.

d. 9-4-1/E-5 (1777) RX BLDG VENT MONITOR DIV I HIGH RAD
9-4-1/E-5 (1778) RX BLDG VENT MONITOR DIV II HIGH RAD
9-4-1/E-4 (1763) RX BLDG VENT MONITOR A HI-HI RAD
9-4-1/E-4 (1780) RX BLDG VENT MONITOR D HI-HI RAD

ANSWER: RO 36

d. 9-4-1/E-5 (1777) RX BLDG VENT MONITOR DIV I HIGH RAD
9-4-1/E-5 (1778) RX BLDG VENT MONITOR DIV II HIGH RAD
9-4-1/E-4 (1763) RX BLDG VENT MONITOR A HI-HI RAD
9-4-1/E-4 (1780) RX BLDG VENT MONITOR D HI-HI RAD

Explanation:

The setpoint for the RX BLDG VENT MONITOR HIGH RAD alarms is 5 mr/hr so both divisions would be in. The RX BLDG VENT MONITOR HI-HI RAD alarms are set at 10 mr/hr so only A and D would have these alarms. This would also initiate a PCIS Group 6.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 37	5537	00	11/19/1999	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	Heating, Ventilation & Air Conditioning

Related Lessons	
COR0010802	OPS HEATING, VENTILATION AND AIR CONDITIONING

Related Objectives	
COR0010802001160D	Predict the consequences a malfunction of the following would have on the Control Room HVAC system: Fire protection

Related References	
2.2.84	HVAC Main Control Room and Cable Spreading Room
10CFR55.41(b)9	

Related Skills (K/A)	
290003.2.1.31	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.(CFR: 41.10/45.12) (4.6/4.3)

QUESTION: RO 37

The plant is operating a 100% power with Control Room Supply Fan SF-C-1A running AND SF-C-1B stopped when a fire occurs in the Cable Spreading Room AND actuates the Smoke Detector in the room exhaust.

Which choice below describes the status of the Control Room Ventilation System for the conditions above?

a. Fire Dampers isolate the Control Room.
Control Room Supply Fan SF-C-1A continues to run.

b. Fire Dampers isolate the Cable Spreading Room.
Control Room Supply Fan SF-C-1A continues to run.

c. Fire Dampers isolate the Control Room.
Control Room Supply Fan SF-C-1A trips.

d. Fire Dampers isolate the Cable Spreading Room.
Control Room Supply Fan SF-C-1A trips.

ANSWER: RO 37

c. Fire Dampers isolate the Control Room.
Control Room Supply Fan SF-C-1A trips.

EXPLANATION:
Smoke detector actuation will close 6 fire dampers to isolate the Control Room and trip the running Control Room Supply Fan. Once the smoke detector is reset, the dampers will reopen and the fan will restart provided control switch positions were not changed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 38	1730	00	08/05/1999	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	2	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022302001030A, COR0022302001150D Residual Heat Removal System

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives
COR0022302001150D Given plant conditions, determine if the following should occur: RHR valve reposition
COR0022302001030A Describe RHR System design feature(s) and/or interlocks which provide for the following: Automatic initiation/injection

Related References
10CFR55.41(b)7

Related Skills (K/A)
219000.K4.09 Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Heat exchanger cooling (3.3/3.4)

QUESTION: 38 1730 (1 point(s))

The plant was operating at near rated power when a small LOCA occurred in the DW.

- Suppression pool temperature has risen to 98°F.
- Reactor water level has lowered to -114 inches.

What is the availability of suppression pool cooling?

a. MO-66, Heat Exchanger Bypass Valve, is sealed open for 5 min
MO-27, LPCI Outboard Injection Valve, is sealed, open for 5 min.

b. MO-66, Heat Exchanger Bypass Valve, is sealed open for 2 min
MO-27, LPCI Outboard Injection Valve, is sealed open for 5 min.

c. MO-66, Heat Exchanger Bypass Valve, is sealed open for 2 min
MO-27, LPCI Outboard Injection Valve, is sealed open for 2 min.

d. MO-66, Heat Exchanger Bypass Valve, is sealed open for 5 min
MO-27, LPCI Outboard Injection Valve, is sealed open for 2 min.

ANSWER: 38 1730
b. MO-66, Heat Exchanger Bypass Valve, is sealed open for 2 min
MO-27, LPCI Outboard Injection Valve, is sealed open for 5 min.

Explanation:

When a LOCA signal occurs these valves are sealed to ensure for the first 2 minutes following a local all RHR injection flow bypasses the RHR HX. Once the timer times out RHR flow can be sent through the HX to allow for Containment Cooling.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 39	9636	00	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	NEW

Topic Area	Description
Systems	RPV level response

Related Lessons
ACD0070307 Thermal Hydraulics

Related Objectives
ACD00703070012600 Describe the factors affecting single and two phase flow resistance.

Related References
10CFR55.41(b)14

Related Skills (K/A)	
295005.AK1.03	Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.8 to 41.10) Pressure effects on reactor level (3.5/3.7)

QUESTION: RO 39

The plant is operating at near rated power when a load reject occurs. What is the immediate effect on reactor water level, if any?

a. No effect
b. Reactor water level will go down rapidly
c. Reactor water level will go up rapidly
d. Reactor water level will slowly lower over the next few minutes.

ANSWER: RO 39

b. Reactor water level will go down rapidly

Explanation:
Due to the rapid collapsing of voids in the core region there is less resistance to flow. Therefore more water will leave the downcomer region and move inside the shroud so the immediate affect is a rapid drop in level of over 35” on the Reactor water level instruments.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 40	9644	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Emergency Operating Procedures	Depress during ATWS

Related Lessons
INT0080618 OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives
INT00806180020400 Using the Cautions provided in the EOP and SAG Flowcharts, explain the bases behind each of the Cautions.

Related References
10CFR55.41(b)10

Related Skills (K/A)
295037.EK1.01 Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.8 to 41.10) Reactor pressure effects on reactor power. (4.1*/4.3*)

QUESTION: RO 40

Given the following conditions:

- ATWS Event in progress
- SLC has been injecting for 3 minutes
- Immediate SCRAM actions complete
- Reactor pressure being maintained by the Turbine Bypass Valves and SRVs

What effect could an inadvertent Reactor cooldown during this event have regarding the plant?

A Cooldown of the Reactor will cause...

a. the MSIVs to close at 835 psig thereby further challenging primary containment

b. SLC effectiveness to decrease due to less boron being dissolved in the Reactor coolant at lower temperatures

c. SLC effectiveness to increase due to more boron sent through the reactor due to increased core flow

d. Reactor power to increase or cause the Reactor to return to criticality

ANSWER: RO 40

d. Reactor power to increase or cause the Reactor to return to criticality

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 41	19285	03	04/03/2007	9/24/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Abnormal/Emergency Procedures	INT0080618, DW Spray with NPSH exceeded

Related Lessons
INT0080618 OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives	
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

Related References
10CFR55.41(b)10

Related Skills (K/A)
295026.EK1.01 Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.8 to 41.10) Pump NPSH. (3.0/3.4)

QUESTION: RO 41

Drywell Sprays, Torus Sprays and Torus Cooling are in service following a LOCA. Several minutes later, the following conditions exist:

- Drywell pressure is 5 psig .
- Drywell temperature is 250°F.
- Torus pressure is 3 psig.
- Torus average water temperature is 180°F.
- Primary containment level is 10 feet.
- RHR loop A system flow is 8000 gpm (ONLY RHR pump A is operating).
- The Control Room Supervisor has directed that operation of RHR and CS pumps shall be maintained within NPSH and Vortex limits.

How is RHR flow affected (if at all) by this direction and why?

RHR flow . . .

a. may continue at current values as no limit is being exceeded.

b. must be reduced since RHR Pump NPSH limit is being exceeded.

c. must be reduced since RHR Pump vortex limit is being exceeded.

d. must be secured because the Drywell Spray Initiation Limit has been exceeded.

ANSWER: RO 41

b. must be reduced since RHR Pump NPSH limit is being exceeded.

The NPSH curve is being exceeded requiring flow to be reduced. (Based on the RHR NPSH curve, it appears that RHR flow will have to be reduced to ~ 6500 gpm, or less, in order to satisfy the NPSH curve again.)

Provide EOP Graphs to student

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 42	133	00	05/01/2000	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Abnormal/Emergency Procedures	INT0320126C0C0400 CNS Abnormal Procedures (RO) Cooling Water

Related Lessons
INT0320126 CNS Abnormal Procedures (RO) Cooling Water

Related Objectives	
INT0320126O0O0100	Given plant condition(s), determine from memory if a manual reactor scram or an emergency shutdown from power is required due to the event(s).

Related References	
2.4TEC	TEC Abnormal

Related Skills (K/A)	
295018.AK2.02	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8) Plant operations (3.4/3.6)

QUESTION: RO 42

The plant is operating at 100% power when the TEC pumps common discharge header ruptures.

What action is required NEXT by procedure if at any time TEC PRESSURE cannot be restored and maintained above 55 psig?

a. Trip all running TEC pumps.

b. Start available Service Water pumps.

c. Scram and concurrently enter Procedure 2.1.5.

d. Perform rapid shutdown per Procedure 2.1.4.1.

ANSWER: RO 42

c. Scram and concurrently enter Procedure 2.1.5.

Procedure 2.4TEC states that "If at any time TEC PRESSURE cannot be restored and maintained above 55 psig: a scram is required and a turbine trip. Answer a is incorrect because this is not the NEXT action required by this condition. Answer b is incorrect because this is not the Next action required by the procedure. Answer d is incorrect because a scram is required with concurrent entry into 2.1.5.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 43	9645	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
INT0070501 OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010800	From memory, state each CNS Safety Limit and discuss the basis for each of the Safety Limits.
INT00705010010900	State the actions which must be performed should a Safety Limit violation occur at CNS.

Related References
10CFR55.41(b)3

Related Skills (K/A)	
295025.EK1.05	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Exceeding safety limits (4.4*/4.7*)

QUESTION: RO 43

The plant is performing the SP 6 .MISC.502 ASME Class 1 System Leakage Test following refueling operations. A miscommunication results in a significant reactor pressure rise. Pressure as read on the Control Room Wide Range Pressure indication is pegged upscale.

The Post Accident Pressure recorders indicate pressure has reached 1350 psig.

Which one of the following is a correct statement with regard to the CNS Safety Limit for Reactor Pressure?

a. Reactor pressure is above the reactor scram setpoint of 1050 psig, reactor pressure must be reduced within 2 hours and all insertable control rods must be inserted immediately.

b. Reactor pressure is above the CNS safety limit of 1250 psig, reactor pressure must be immediately reduced and all insertable control rods must be inserted with in 2 hours.

c. Reactor pressure is above the CNS safety limit of 1337 psig, reactor pressure must be reduced within 2 hours and all insertable control rods must be inserted with in 2 hours

d. Reactor Pressure is below the CNS safety limit of 1375 psig, all insertable control rods must be inserted immediately and 2.1.5 and EOP-1A should be entered.

ANSWER: RO 43

c. Reactor pressure is above the CNS safety limit of 1337 psig, reactor pressure must be reduced within 2 hours and all insertable control rods must be inserted with in 2 hours.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 44	9652	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
COR0020302 OPS CONTAINMENT COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives
COR0020302001060G Describe the interrelationships between PCIS and the following: SDC/RHR COR0022302001070A Given a specific RHR system malfunction, determine the effect on any of the following: Reactor parameters (level, pressure, temperature)

Related References
10CFR55.41(b)7

Related Skills (K/A)
295031.AK3.01 Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following:(CFR: 41.7 / 45.8) Reactor pressure (3.8/3.9)

QUESTION: RO 44

The plant is shutdown for a refueling outage after running continuously for 432 days.

- Shutdown cooling has been in service for the last 30 minutes using RHR subsystem A with RHR Pump A running.
- RHR Subsystem B is tagged out for maintenance on RHR Pump B and RHR Pump D.
- RPS ‘B’ is lost due to maintenance activities in the area of the RPS MG set.
- One minute later an inadvertent LPCI initiation signal occurs on low reactor water level.

Which of the following describe the affect this will have:

a. RHR-MO-25A and RHR-MO-27A are open. RHR pumps A and C are injecting. Reactor level will increase and reactor pressure and temperature will lower rapidly.

b. RHR-MO-25A is open and RHR-MO-27A is open. RHR pump A is idle and RHR pump C is running. Reactor level will increase and reactor pressure and temperature will lower rapidly.

c. RHR-MO-25A is open and RHR-MO-27A is open. RHR pump A is tripped and RHR Pump C is idle. Reactor Pressure will begin to rise due to the loss of SDC.

d. RHR-MO-25A is closed and RHR-MO-27A is open. RHR pump A is running and RHR pump C is idle. Reactor Pressure will begin to rise due to the loss of SDC.

ANSWER: RO 44

c.   RHR-MO-25A is open and RHR-MO-27A is open. RHR pump A is tripped and RHR Pump C is idle. Reactor Pressure will begin to rise due to the loss of SDC.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 45	3173	01	09/08/1999	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Abnormal/Emergency Procedures	INT 0320134C0C0300 CNS Abnormal Procedures (RO) - Fire

Related Lessons
INT0320134 OPS CNS Abnormal Procedures (RO) - Fire

Related Objectives
10CFR55.41(b)10

Related References	
5.4FIRE-SD	Fire Induced Shutdown From Outside Control Room

Related Skills (K/A)	
295016.AK3.01	Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6) Reactor SCRAM (4.1*/4.2*)

QUESTION: RO 45

The Plant is operating at 100% power

- A fire has been confirmed in the Auxiliary Relay Room AND the Fire Brigade is on the scene
- Narrow Range Reactor level is 35" AND Drywell pressure is 0.26 psig
- High Pressure Coolant Injection AND Core Spray B initiate AND align for injection
- The actions of 5.1INCIDENT AND 5.4POST-FIRE are being performed
- The Control Room Supervisor enters 5.4Fire-SD to commence shutdown from outside the Control Room procedure

In accordance with 5.4Fire-SD, which statement below describes the Subsequent Operator Actions required for these conditions and reason?

a. The Control Room is to be immediately evacuated to ensure personnel safety.

b. Prior to leaving the Control Room, the Reactor must be scrammed to ensure the reactor will remain shutdown under all conditions without boron.

c. IF time permits, Scram the Reactor AND verify ALL rods in prior to evacuating the Control Room to ensure the reactor will remain shutdown under all conditions without boron.

d. Prior to leaving the Control Room, Station personnel must be informed of the Control Room evacuation to ensure the station can contact the Shift crew via cell phones.

ANSWER: RO 45

b. Prior to leaving the Control Room, the Reactor must be scrammed to ensure the reactor will remain shutdown under all conditions without boron.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 46	19222	00	05/27/2002	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Abnormal/Emergency Procedures	COR0010102, AC Electrical Distribution (2002 NRC Exam)

Related Lessons
COR0010102 AC Electrical Distribution INT0320134 OPS CNS Abnormal Procedures (RO) - Fire

Related Objectives
INT0320134E0E0100 Given plant condition(s), determine from memory any automatic actions listed in the applicable Abnormal/Emergency Procedure(s) which will occur due to the event(s). COR0010102001080A Predict the consequences of the following on plant operation: Loss of Normal and Startup transformers

Related References
10CFR55.41(b)7

Related Skills (K/A)
295005.AK3.02 Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6) Recirculation pump downshift/trip: Plant-Specific (3.4/3.5)

QUESTION: RO 46

The plant is operating at 100% power with a normal electric plant lineup when an oil leak occurs on the Normal Transformer. The oil catches on fire. The fire brigade responds and attempts to extinguish the blaze. The oil and the fire is contained to the Normal Transformer area but it can not be put out.

What is the potential effect of this fire?

a. The loss of one (1) Recirculation pump only.

b. A main generator trip AND Reactor scram only.

c. The loss of BOTH Recirculation pumps AND main generator trip AND Reactor scram.

d. The loss of only one (1) Recirculation pump AND main generator trip AND Reactor scram.

ANSWER: RO 46

d. The loss of only one (1) Recirculation pump AND main generator trip AND Reactor scram.

The NORMAL transformer will trip due to the fire, thus resulting in loss of the Normal Transformer. The Main Generator will trip resulting in a reactor scram and the loss of the RR Pump that is being power from the normal transformer.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 47	1063	00	06/22/1999	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022202, REACTOR RECIRCULATION

Related Lessons
COR0021502 OPS NUCLEAR BOILER INSTRUMENTATION COR0022202 REACTOR RECIRCULATION SKL0124222 OPS REACTOR RECIRCULATION SYSTEM

Related Objectives
COR0021502001060K Given a specific NBI malfunction, determine effect on any of the following: Core flow/Jet Pump monitoring COR0022202001060B Given a specific Reactor Recirculation system or the Recirculation Flow Control system malfunction, determine the effect on any of the following: Core Flow (normal and reduced forced flow conditions) SKL012422200A030E Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: Core flow

Related References
2.2.68.1 Reactor Recirculation System Operations 10CFR55.41(b)6

Related Skills (K/A)
295001.AK3.03 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.5 / 45.6) Idle loop flow (2.8/2.9)

QUESTION: RO 47

Given the following conditions:

- The "B" Recirculation Pump has tripped.
- MO-53B, "B" Recirculation Pump discharge valve was closed and is now open.
- LOOP B JET PUMP FLOW (FI-92B) indicates 2 Mlbm/hr.
- LOOP A JET PUMP FLOW (FI-92A) indicates 35 Mlbm/hr.
- Annunciator E-7 on Panel 9-4-3, RECIRC LOOP B OUT OF SERVICE is NOT alarming.

What is the expected value for indicated Total Core Flow as indicated on Panel 9-5 Recorder dPR/FR-95 AND what is Actual Core Flow?

Indicated total Core Flow will be _______ and *actual* core flow will be ________.

a. 33 Mlbm/hr, 33 Mlbm/hr

b. 33 Mlbm/hr, 37 Mlbm/hr

c. 37 Mlbm/hr, 33 Mlbm/hr

d. 37 Mlbm/hr, 37 Mlbm/hr

ANSWER: RO 47

c. 37 Mlbm/hr, 33 Mlbm/hr

EXPLANATION OF ANSWERS: c. Correct. With one Recirc Pump out of service, a reverse flow will exist through the idle Jet Pumps but the Jet Pump Flow instrumentation will indicate a positive flow. Annunciator E-7 not in alarm indicates Core Flow circuitry is not functioning properly for single loop (i.e. The Loop Jet Pump flows are being added vice subtracted). Total Core Flow will indicate 37 on dPD/FR-95. Since reverse flow exists in the idle loop, Actual core flow will be the difference between Loop A & B Jet Pump flows. a,b,d. With the Core Flow summing circuit malfunctioning, indicated Total Core Flow will be the sum of Loop A & B Jet Pump flows (37) and Actual Core Flow will be the difference between Loop A & B Jet Pump Flows (33).

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 48	23374	01	07/06/2006	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	3	Multiple Choice	BANK

Topic Area	Description
None	COR0010102, For the given plant condition, which of the following will trip?

Related Lessons	
COR0010102	AC Electrical Distribution

Related Objectives	
COR0010102001130K	Predict the consequences of the following events on the AC Electrical Distribution System: Degraded system voltages

Related References	
2.2.18	4160V Auxiliary Power Distribution System
2.2.19A	480 VAC Auxiliary Power Checklist
10CFR55.41(b)7	

Related Skills (K/A)
700000.AA1.05 Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8) Engineered safety features (3.9/4.0)

QUESTION: RO 48

The Plant is in Mode 1 at full power with the following conditions:

- RHR A pump has been placed in service for suppression pool cooling
- Service water pump A, B, and D in service
- CRD pump A in service

A degrading voltage condition starts occurring on 4160 VAC Bus 1F which results in annunciator C-1/A-7, 4160V BUS 1F LOW VOLTAGE.

For the given plant condition, which motors will trip immediately?

a. Service Water pump A; CRD pump A

b. RHR pump A; Service Water Pump A

c. Station Air Compressor 1A; CRD pump A

d. Station Air Compressor 1A; RHR pump A

ANSWER: RO 48

b. RHR pump A; Service Water Pump A

Explanation: is correct. Breaker 1FA trips open due to degraded voltage (Relay 27X12/1F) on 4160V bus 1F. After breaker 1FA trips open, undervoltage on 4160V bus 1F causes trips of Service Water and RHR pumps and - at the same time -- signals breaker 1FS to close, powering bus 1F from the Emergency Station Service Transformer -- after large motors have been tripped.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 49	23317	01	06/20/2006	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022102, What is the status of RPS five (5) seconds later directly due to manipulating the Reactor Mode Switch? (ILT 2006 AUDIT EXAM)

Related Lessons	
COR0022102	REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001050D	Briefly describe the following concepts as they apply to RPS: Mode switch position

Related References
10CFR55.41(b)7

Related Skills (K/A)	
295006.AA1.01	Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6) RPS. (4.2*/4.2*)

QUESTION: RO 49

While at 100% power, the Reactor Mode switch is placed in SHUTDOWN.

No other operator action is taken.

What is the status of the following RPS indications five (5) seconds later directly due to manipulating the Reactor Mode Switch?

a. RX SCRAM CHANNEL A/B annunciators in alarm.
MANUAL SCRAM pushbuttons red backlighting ON.

b. RX SCRAM CHANNEL A/B annunciators in alarm.
MANUAL SCRAM pushbuttons red backlighting OFF.

c. RX SCRAM CHANNEL A/B annunciators not in alarm.
MANUAL SCRAM pushbuttons red backlighting ON.

d. RX SCRAM CHANNEL A/B annunciators not in alarm.
MANUAL SCRAM pushbuttons red backlighting OFF.

ANSWER: RO 49

a. RX SCRAM CHANNEL A/B annunciators in alarm.
MANUAL SCRAM pushbuttons red backlighting ON.

Explanation: With the Reactor Mode switch in SHUTDOWN it de-energizes the manual scram channel causing the manual scram pushbuttons to backlight and the RX SCRAM alarms will alarm due to deenergizing A3 and B3 manual scram channels.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 50	15129	01	06/19/2007	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Electrical	COR0020702, DC Electrical Distribution

Related Lessons	
COR0020702	OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001060C	Describe the interrelationship between the DC Electrical Distribution System and the following: Battery charger and battery
COR0020702001080C	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Systems with DC components (i.e., valves, motors, solenoids, etc.)

Related References	
IV.VIII	Electrical Power Systems

Related Skills (K/A)
295003.AA1.04 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.7 / 45.6) D.C. electrical distribution system (3.6/3.7)

QUESTION: RO 50

A complete loss of AC power occurred 2 hours ago and continues to affect the plant.

Based upon CNS Station Black Out (SBO) Analysis, the 250VDC power system batteries is/are?

a. Expected to operate normally for the next 2 hours.

b. Past the failure point and individual cells are expected undergo cell reversals .

c. At the high load operating time and will start to fail within the next hour.

d. Capable of operating for another 6 hours provided that compensatory measures are accomplished per procedure.

ANSWER: RO 50

a. Expected to operate normally for the next 2 hours.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 51	9653	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
COR0020302 CONTAINMENT

Related Objectives
COR0020302001230A Predict the consequences of a malfunction of the following on the Primary containment: Drywell cooling.

Related References
10CFR55.41(b)5

Related Skills (K/A)
295028.EA2.05 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13) Torus/suppression chamber pressure: Plant-Specific (3.6/3.8)

QUESTION: RO 51

The plant is operating near rated power in the month of August.

REC-MO-702 REC Supply to the Drywell fails shut and cannot be reopened

How will this affect the torus over the next 15 min?

Torus Level will ______ and Torus Pressure will ________.

a. Lower; Lower

b. Lower; Rise

c. Rise; Lower

d. Rise; Rise

ANSWER: RO 51

d. Rise; Rise

Explanation: Due to the loss of REC to the DW, DW temperature and pressure will increase. This will result in the water in the downcomers being forced into the torus area and torus level will go up. Along with this Torus air space will be reduced due to the rising water level and Torus pressure will rise.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 52	24765	0	02/16/2008	09/18/2008	License Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	2	Multiple Choice	BANK

Topic Area	Description
None	HOLD NRC 2009

Related Lessons
ACD0023111 Batteries

Related Objectives
ACD00231110010000 Describe the operating characteristics of a lead-acid battery to include methods of voltage production, state of charge, and hazards associated with storage batteries.

Related References
10CFR55.41(b)5

Related Skills (K/A)
295004.AA2.03 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.10 / 43.5 / 45.13) Battery voltage (2.8/2.9)

QUESTION: RO 52

The plant was operating in a normal full power line-up when multiple failures resulted in a total loss of AC power (Station Blackout). Three hours into the event HPCI is the only injection source available.

Therefore the crew decided that the use of HPCI is required. HPCI is placed in operation for 30 minutes.

- For the first 25 minutes of HPCI operation division II 250 VDC voltage dropped from 264 VDC to 240 VDC.
- In the last 5 minutes of HPCI operation battery voltage dropped from 240 VDC to 198 VDC.

How much Division II battery capacity remains?

a. Very little battery capacity remains.

b. More than half of the battery capacity remains.

c. 75%

d. 82.5%

ANSWER: RO 52

a. Very little battery capacity remains.

Explanation:

Battery capacity remaining is a function of the amp-hours that the battery has supplied. Voltage is not an indicator of battery capacity. Battery voltage is influenced by too many things to provide an indication of remaining capacity. Only amp-hours out of the battery and electrolyte specific gravity provide an accurate picture of remaining capacity. In addition battery voltage does not decrease linearly but once the battery is capacity is exhausted voltage drops rapidly.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 53	6041	0	01/25/2000	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Administrative	GEN0030401 Emergency Plan for Licensed Operators

Related Lessons	
GEN0030401	Emergency Plan for Licensed Operators

Related Objectives	
GEN0030401D0D0600	State the Protective Action Guides used to determine the need to evacuate the general public.

Related References	
5.7.20	Protective Action Recommendations
5.7.1	Emergency Classification
10CFR55.41(b)12	

Related Skills (K/A)
295038.EA2.03 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.10 / 43.5 / 45.13) Radiation levels (3.5*/4.3*)

QUESTION: RO 53

What are the Protective Action Guideline (PAG) values used to determine the need to evacuate the general public?

a. 1R exposure or 25 mr/hr exposure rate.

b. 1 REM TEDE or 5 REM CDE Thyroid.

c. 5R exposure or 25 mr/hr exposure rate.

d. 5 REM TEDE or 25 REM CDE Thyroid.

ANSWER: RO 53

b. 1 REM TEDE or 5 REM CDE Thyroid

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	12518	0	3/7/2001	3/8/2007	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	DC Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives
COR0020702001080D Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Battery Chargers

Related References
10CFR55.41(b)7

Related Skills (K/A)
295004.2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7) (4.1/4.1)

QUESTION: RO 54

The plant is at near rated power when the following occur:

- Annunciator C-4/C-7, 125 VDC BATT CHARGER 1B TROUBLE alarms
- CRT alarm message indicates:
  - (3765) 125V DC BATTERY CHARGER 1B DC VOLTAGE HIGH (in and reset)
  - (3762) 125V DC BATTERY CHARGER 1B AC VOLTAGE FAILURE.
  - (3764) 125V DC BATTERY CHARGER 1B DC VOLTAGE LOW

The following 125 VDC indications are observed:

- 125 VDC Bus 1B indicates approximately 120 volts and stable
- 125 VDC Battery 1B indicates approximately 75 amps OUT and stable
- 125 VDC Charger 1B indicates 0 amps

What is the position (open or closed) of the 125V charger 1B AC input and DC output breakers and why?

a. AC input breaker tripped open due to a loss of AC power to the chargers.
DC output breaker remains closed.

b. AC input breaker tripped open due to battery charger 1B DC voltage high.
DC output breaker remains closed.

c. AC input breaker **AND** DC output breaker tripped open due to a loss of AC power to the chargers.

d. AC input breaker **AND** DC output breaker tripped open due to battery charger 1B DC voltage high.

ANSWER: RO 54

b. AC input breaker tripped open due to battery charger 1B DC voltage high.
DC output breaker remains closed.

The AC input breaker has tripped open due to battery charger 1B DC voltage high. DC output over voltage causes the AC input breaker on a 125V CHARGER to trip. The DC output breaker does NOT automatically trip open. Neither breaker automatically trips open due to loss of AC power to the chargers.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 55	20809	00	06/02/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	2	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070505, Reactor Pressure TS3.4.10

Related Lessons
INT0070505 CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.

Related References	
3.4.10	Reactor steam dome pressure
10CFR55.41(b)10	

Related Skills (K/A)	
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2) (4.0/4.7)

QUESTION: RO 55

The plant was operating at near rated power when a lightning strike to the ERP results in a DEH transient that causes DEH pressure setpoint to shift to manual at a higher pressure. The following conditions were noted after conditions stabilized:

- Reactor power is 99%.
- Annunciator 9-5-2/F-2 Reactor High Pressure is alarming.
- Reactor pressure is 1025 psig and steady.
- Generator gross load is 786 MW.
- Generator reactive load is +100 MVARs.
- DEH is in Mode IV with the governor valves controlling pressure.

What action(s) is/are required?

a. Immediately scram and concurrently enter Procedure 2.1.5.

b. Immediately reduce generator gross load to less than 780 MW.

c. Reduce reactor pressure to less than 1004 psig within 15 minutes.

d. Immediately reduce generator reactive load to less than +50 MVARs.

ANSWER: RO 55

c. Reduce reactor pressure to less than 1004 psig within 15 minutes.

If pressure exceeds 1004 psig during a DEH transient, pressure must be restored and maintained < 1004 psig to ensure OPL 3, Transient Protection Parameters Verification for Reload Licensing Analysis, is met.

In modes 1 and 2 reactor steam dome pressure is required to be less than 1020 psig. TS 3.4.10 requires that pressure be reduced to less than 1020 psig within **15 minutes**.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 56	2369	02	10/06/2005	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070505, CNS Technical Specifications 3.4, Reactor Coolant System (RCS)

Related Lessons
INT0070505 CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050010600	From memory, in MODE 3 with reactor steam dome pressure less than or equal to the shutdown cooling permissive pressure, state the actions required in less than one hour for one or two RHR shutdown cooling subsystems inoperable (LCO 3.4.7).

Related References	
3.4.7	Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown
10CFR55.41(b)10	

Related Skills (K/A)	
295021.2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2) (3.2/4.2)

QUESTION: RO 56

Given the following plant conditions:

- Reactor shutdown and cooldown in progress
- Reactor pressure is 48 psig
- Reactor temperature is 275°F
- RR Pump A is in Operation
- The "B" loop of Shutdown Cooling (SDC) in operation
- "A" loop of SDC is declared inoperable
- Turbine Building Station Operator reports "B" RHR Pump Breaker has been racked OUT.

What action is required per Technical Specifications *<u>in less than one hour</u>* and why?

a. Stabilize reactor coolant temperature; to prevent exceeding the allowable Heat Up Rate.

b. Verify one (1) recirculation pump in operation; to ensure forced circulation through the core.

c. Verify an alternate method of decay heat removal is available; this re-establishes backup decay heat removal capabilities.

d. Ensure an alternate method of reactor coolant circulation is in operation; to ensure forced circulation through the core.

ANSWER: RO 56

c. Verify an alternate method of decay heat removal is available; this re-establishes backup decay heat removal capabilities.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 57	19899	01	03/10/2003	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Administrative	COR0012102001030C, Refueling

Related Lessons	
COR0012102	Refueling

Related Objectives	
COR0012102001030C	Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Core reactivity level

Related References	
10.25	Refueling - Core Unload, Reload, and Shuffle
10CFR55.41(b)10	

Related Skills (K/A)	
295023.AK1.02	Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: (CFR: 41.8 to 41.10) Shutdown margin. (3.2/3.6)

QUESTION: RO 57

A core reload is in progress.

- SRM "B" count rate is 50 cps
- Fuel Pool water temperature is 95°F .
- A fuel bundle is being lowered into the core and is just passing through the top guide.
- SRM "B" count rate rises to 500 cps.

What action is required to be performed?

a. Immediately terminate fuel loading.

b. Continue to insert the bundle normally. If the SRM reaches 5 "doubles," terminate fuel loading.

c. Insert the bundle half way into the core, then stop and monitor the count rate. Remove the fuel bundle from the core if the SRM reaches 5 "doubles."

d. Slowly lower the bundle into the core by moving in six (6) inch increments, stopping to monitor SRM count rates at each increment. Terminate fuel loading if the SRM reaches 5 "doubles."

ANSWER: RO 57

a. Immediately terminate fuel loading.

Answer source: 10.25

Distractors:

b. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.
c. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.
d. Procedure 10.25 requires fuel handling be terminated if unexpected rise on SRMs is noted. The maximum count rate expected during a fuel shuffle is 100 cps.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 58	2244	01	07/21/2004	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Academics	COR0011302, MAIN GENERATOR AND AUXILIARIES

Related Lessons	
COR0011302	OPS MAIN GENERATOR AND AUXILIARIES
COR0030305	Motors & Generators (GP)

Related Objectives	
COR00303050002000	Describe the effect of changing the excitation of a generator connected to the grid.
COR0011302001080C	Predict the consequences of the following on the Main Generator and Auxiliaries: Electrical distribution malfunction
COR00303050001900	Describe how to adjust generators connected in parallel for variations in shared load.
COR0011302001080I	Predict the consequences of the following on the Main Generator and Auxiliaries: Grid instabilities
COR0011302001140D	Briefly explain the following concepts as they apply to the Main Generator: Reactive load

Related References
NONE

Related Skills (K/A)	
700000.AA1.03	Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8) Voltage regulator controls (3.8/3.7)

QUESTION: RO 58

The plant is operating at rated power with +50 MVAR on the Main Generator (voltage regulator in automatic). Grid problems result in slowly lowering 345 KV voltage.

If no operator action is taken, how will this lowered voltage affect the main generator?

Main Generator MVARs ...

a. rise and generator field amps rise.

b. rise and generator field amps lower.

c. lower, then become negative and generator field amps rise.

d. lower, then become negative and generator field amps lower.

ANSWER: RO 58

a. rise and generator field amps rise.

As grid voltage lowers, main generator terminal voltage will also lower. The main generator voltage regulator will raise field amps in an attempt to maintain the voltage setpoint (minus the value for reactive droop). As a result, Generator MVARs will rise and become more positive.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 59	19224	00	05/31/2002	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT00806170010100, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Lessons
INT0080617 OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives
INT00806170010100 List the entry conditions to Flowchart 5A (including the radioactivity release path) and briefly explain each.

Related References
NONE

Related Skills (K/A)
295035.EK1.02 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10) Radiation Release (3.7/4.2*)

QUESTION: RO 59

Why is entry into EOPs required if the Reactor Building dP cannot be maintained negative?

This reactor building (RB) dP is an indication that . . .

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

b. the continued operability of equipment needed to carry out EOP actions may be compromised.

c. radioactivity is being released to the environment when the ventilation system should have automatically isolated.

d. an indication that water from a primary system (or from a primary to secondary system leak) may be discharging into the secondary containment.

ANSWER: RO 59

a. an uncontrolled, unmonitored release of radioactivity to the environment could exist.

Distractors:

b. This is the basis for the high temperature entry.
c. This is the basis for the high Rx bldg exhaust radiation level.
d. This is the basis for the entry on radiation above Max Normal Operating Level.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 60	24784	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	2	Multiple Choice	NEW

Topic Area	Description
NEW	HOLD 2009 NRC

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001080A	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation
COR0022802001130A	Given plant conditions, determine if any of the following should occur: SGT automatic initiation

Related References
10CFR55.41(b)9

Related Skills (K/A)	
295033.EK2.02	Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: (CFR: 41.7 / 45.8) Process radiation monitoring system (3.8/4.1)

QUESTION: RO 60

The plant is at 90% when a steam line break occurs. Current conditions are:

- A reactor scram is inserted and all control rods fully insert.
- Temperatures in the Reactor Building are 214°F and rising.
- Reactor Building (RB) ventilation exhaust radiation levels are 25 mr/hr and annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD is alarming.
- Reactor water level is 25"(NR).
- Drywell pressure is 0.75 psig.

What is the status of the release from the reactor building?

a. RB HVAC is operating and the release is monitored via the RB ventilation exhaust radiation monitors.

b. RB HVAC is not operating and the release is unmonitored due to normal RB leakage paths.

c. SGT is operating and the release is monitored via the ERP Kaman.

d. SGT is operating and the release is unmonitored since SGT does not have radiation monitors.

ANSWER: RO 60

c. SGT is operating and the release is monitored via the ERP Kaman.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 61	2140	01	05/15/2002	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	COR0022602, Rod Worth Minimizer

Related Lessons	
COR0022602	OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022602001040A	Describe the RWM design features and/or interlocks that provide for the following: System initialization
COR0022602001040G	Describe the RWM design features and/or interlocks that provide for the following: System bypass
COR0022602001080B	Predict the consequences of the following items on the RWM: Out of sequence rod movement
COR00226020011200	State the reason for bypassing the RWM following an incomplete scram.

Related References	
4.2	Rod Worth Minimizer
10CFR55.41(b)10	

Related Skills (K/A)	
295015.AK3.01	Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM: (CFR: 41.5 / 45.6) Bypassing rod insertion blocks (3.4/3.7)

QUESTION: RO 61

The plant was operating at power when a transient occurred that resulted in a reactor scram signal.

- Not all control rods inserted as required.
- Reactor power is oscillating between the APRM downscale alarm and 10%.
- "Failure To Scram" actions are being performed.

How is RWM/RMCS manipulated during these conditions and why?

a. Manually bypass the RWM to allow rod insertion with RMCS.

b. Use EMERGENCY NOTCH OVERRIDE SWITCH to insert control rods in order to bypass RWM insert blocks.

c. Depress the RWM INITIALIZE push button on the IDT keyboard to initialize the RWM to the current rod pattern to allow rod insertion.

d. Use either EMERGENCY NOTCH OVERRIDE SWITCH or ROD MOVEMENT CONTROL SWITCH because the RWM does not enforce rod blocks at this power.

ANSWER: RO 61

a. Manually bypass the RWM to allow rod insertion with RMCS.

Answer a is correct. At this power level the RWM enforces insert rod blocks preventing rod insertion with RMCS. Bypassing the RWM allows rod insertion with RMCS. Answer b is incorrect because the RWM enforces insert block regardless of whether the Emergency Notch Override Switch or the Rod Movement Control Switch is used. Answer c is incorrect as this action is not allowed by 5.8.3. Answer d is incorrect because the RWM does enforce rod blocks at this power.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 62	24782	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
None	HOLD 2009 NRC

Related Lessons
INT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives
INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References
10CFR55.41(b)10

Related Skills (K/A)
295013.AA1.01 Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE : (CFR: 41.7 / 45.6) Suppression pool cooling (3.9/3.9)

QUESTION: 62 24782 (1 point(s))

The following plant conditions exist:

- A LOCA has occurred.
- Torus pressure is 3 psig and increasing.
- Reactor pressure is 800 psig and decreasing.
- Reactor water level is +15 inches and increasing.
- HPCI is the only system injecting into the reactor.
- Suppression Pool temperature is 100°F and increasing.
- Suppression Pool level is 12.9 feet and increasing.

What action is required?

a. verify both RHR systems are aligned for the LPCI Injection with pumps running and injection valves open.

b. One RHR subsystem should be aligned for Drywell Sprays.

c. Place one loop of RHR in Suppression Pool Cooling and align the other loop for Drywell Sprays.

d. Place one loop of RHR in Suppression Pool Cooling and align the other loop for Torus Sprays.

ANSWER: RO 62

d. Place one loop of RHR in Suppression Pool Cooling and align the other loop for Torus Sprays.

Provide student with EOPs and EOP Graphs with Entry conditions and Cautions blanked out.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 63	11856	03	10/03/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080613, EOP 3A, ED on 280°F outside DWSIL

Related Lessons
INT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives
INT00806130011100 Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References
10CFR55.41(b)10

Related Skills (K/A)
295012.AA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Drywell temperature (3.8/3.9)

QUESTION: RO 63

An accident has occurred with the following conditions:

- Reactor pressure is 590 psig AND lowering slowly.
- Corrected Fuel Zone level is -110" AND stable.
- Drywell pressure is 4.5 psig AND rising slowly.
- Drywell temperature is 281° F AND rising at 2° F per minute.
- PC water level is 12.5 feet AND stable.
- Torus water temperature is 205° F AND rising at 1° F per minute.
- Torus pressure is 3.3 psig AND rising slowly.
- Torus sprays are in service.

What action(s) is/are to be performed by the crew at this time?

a. Spray the Drywell only.

b. Emergency depressurize only.

c. Spray the drywell and emergency depressurize.

d. Terminate Torus Spray and emergency depressurize.

ANSWER: RO 63

b. Emergency depressurize only.

(Unsafe region of DWSIL and DW temp cannot be restored and maintained below 280°F, no requirement to terminate torus spray, DW pressure is > 2.5 psig [the actual setpoint for spray permissive]). No valid PC pressure given to Terminate Torus Spray.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 64	8937	00	08/15/2000	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080610, Why Inhibit ADS during ATWS?

Related Lessons
INT0080610 OPS EOP FLOWCHART 7A - RPV LEVEL (FAILURE-TO-SCRAM)

Related Objectives	
INT00806100010200	State the basis for inhibiting ADS during failure to scram transients.
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.

Related References
10CFR55.41(b)10

Related Skills (K/A)	
295008.2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13) (3.8/4.3)

QUESTION: RO 64

Why is the Automatic Depressurization System (ADS) inhibited in EOP-7A, "RPV Level (Failure-to-Scram)"?

a. Uncontrolled depressurization will put significant energy into the suppression pool before it is necessary.

b. Depressurization could drive plant conditions into the UNSAFE region of the RPV Saturation Temperature curve.

c. Depressurization below the shutoff head of the low pressure injection systems may cause large positive reactivity additions.

d. Uncontrolled depressurization under these conditions would cause a large loss of RPV inventory, lowering reactor water level below the top of active fuel.

ANSWER: Ro 64

c. Depressurization below the shutoff head of the low pressure injection systems may cause large positive reactivity additions.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 65	16445	01	02/21/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080617, FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEA

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
5.8	Emergency Operating Procedures (EOPs)
10CFR55.41(b)10	

Related Skills (K/A)
NONE

QUESTION: RO 65

The plant is at 50% when a RCIC steam line break occurs. Current conditions are:

- RCIC cannot be isolated.
- RCIC temperatures in the NE Quad are 225°F and rising.
- Reactor Building (RB) ventilation exhaust radiation levels are 25 mr/hr and Annunciator 9-4-1/E-4 RX BLDG VENT HI-HI RAD is alarming.
- RCIC/Core Spray Pump Room radiation levels are 300 mr/hr (by survey) and rising.
- A reactor scram is inserted and all control rods fully insert.
- Reactor water level is 25"(NR).
- Drywell pressure is 0.75 psig.

Ten (10) minutes later, Annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD clears.

What is required for RB HVAC and why?

a. Do NOT restart RB HVAC until RP ensures normal radiation levels.

b. Restart RB HVAC to help restore secondary containment parameters.

c. Do NOT restart RB HVAC as EOP-1A requires a PCIS Group 6 isolation.

d. Restart RB HVAC to ensure all radioactive discharges are elevated.

ANSWER: RO 65

b. Restart RB HVAC to help restore secondary containment parameters.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 66	5899	02	03/27/1996	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	4	Multiple Choice	Bank

Topic Area	Description
Systems	IOA for RR Pump Speed increase

Related Lessons	
COR0099900	NRC Licensed Personnel Requalification Written Exams
COR0023202	OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR00999000000100	Open Reference Question

Related References
10CFR55.41(b)10

Related Skills (K/A)	
2.1.2	Knowledge of operator responsibilities during all modes of plant operation. (CFR: 41.10 / 45.13) (4.1/4.4)

QUESTION: RO 66

Following control rod withdrawal raising rod line to 101%, reactor power is being raised from 80% to 100% using recirculation loop flow.

- After raising speed with the manual controller, "A" Reactor Recirc Pump speed continues to rise.
- Depressing the SCOOP TUBE LOCKOUT pushbutton fails to stabilize core flow.

Select the action that is required next:

a. Trip "A" Reactor Recirc pump, and monitor the Power to Flow Map for entry into the instability region.

b. Reduce reactor power by inserting control rods.

c. Dispatch an operator to locally operate "A" Reactor Recirc Motor Generator Set scoop tube, and balance recirc flow.

d. Initiate an emergency shutdown from power, due to entering the Scram Region of the Power to Flow Map.

ANSWER: RO 66

a. Trip "A" Reactor Recirc pump, and monitor the Power to Flow Map for entry into the instability region.

FOILS:

b. Inserting rods would be required action to exit the instability region but the plant is not operating in this region.

c. This is a subsequent action, immediate actions must be performed first.

d. Only required if both RR pumps trip while > 1% power, or in the Scram region of the Power to Flow Map

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 67	24785	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3		NEW

Topic Area	Description
None	HOLD 2009 NRC

Related Lessons
INT0070501 OPS Introduction to Technical Specifications INT0231002 Pre-Outage Industry Events

Related Objectives
INT0070501001030A From memory, define the following terms: Core Alteration

Related References
10CFR55.41(b)10

Related Skills (K/A)
2.1.36 Knowledge of procedures and limitations involved in core alterations. (CFR: 41.10 / 43.6 / 45.7) (3.0/4.1)

QUESTION: RO 67

The plant is in a day 18 of a refueling outage with the following items to be performed on the shift. Which of these would be a CORE ALTERATION?

a. Install a control rod blade into an empty cell.

b. Drive an Intermediate Range Monitor detector to full in.

c. Move a new fuel bundle from the Spent Fuel Pool to the Reactor Vessel.

d. Insert the LPRM Instrument Handling Tool below the top guide.

ANSWER: RO 67

c. Move a new fuel bundle from the Spent Fuel Pool to the Reactor Vessel.

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 68	10832	02	09/20/2005	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	BANK

Topic Area	Description
Administrative	INT0320101, Rules for quality procedures

Related Lessons	
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010100A010A	Discuss the following as described in Administrative Procedure 0-CNS-02, Site Work Practices: Care and Correction of Quality Documents.

Related References	
0-CNS-02	Site Work Practices
10CFR55.41(b)10	

Related Skills (K/A)	
2.2.6	Knowledge of the process for making changes to procedures. (CFR: 41.10 / 43.3 / 45.13) (3.0/3.6)

QUESTION: RO 68

A new, one-time use procedure was written to test a safety related heat exchanger on the emergency diesel generator for biological corrosion (MIC). The procedure has been classified as a quality document.

During your review to perform this procedure the following items are noted. Which of these would violate CNS guidelines for quality procedures?

a. It contains a non-quality segment.

b. Part of the quality portion has been hand-written.

c. A step was deleted by single lining out with black ink, with initials and date placed next to the correction.

d. A valve number was corrected using correction tape with the new number typed on it, initialed and dated.

ANSWER: RO 68

d. A valve number was corrected using correction tape with the new number typed on it, initialed and dated.

a. Quality documents are allowed to contain non-quality segments (step 3.6).
b. Quality documents may be hand written (3.5).
c. Describes the proper way to change a quality document.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 69	4049	01	10/01/1992	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Administrative	INT0320101, CNS Administrative Procedures Volume 0, Administrative Proce

Related Lessons	
OTH0159118	12.5 KV Distribution System
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)
SKL0080305	Tagout General
COR0099900	NRC Licensed Personnel Requalification Written Exams

Related Objectives	
INT032010100C010C	Discuss the following as described in Administrative Procedure 0-CNS-52, Control of Switchyard and Transformer Yard Activities at CNS: Methods of Configuration Control
INT032010100C010B	Discuss the following as described in Administrative Procedure 0-CNS-52, Control of Switchyard and Transformer Yard Activities at CNS: General work requirements
SKL00803050010100	Identify the procedures governing the tagout program as CNS.
COR00999000000100	Open Reference Question

Related References	
2.2.90	12.5 KV System
10CFR55.41(b)10	

Related Skills (K/A)	
2.2.17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (CFR: 41.10 / 43.5 / 45.13) (2.6/3.8)

QUESTION: RO 69

The 13.8 to 12.5 KV transformer in the switchyard is being tagged out for replacement.

Which one of the following requirement sets apply?

a. A switching order must be hung by a qualified switchman and the clearance order tags must be hung by an operator.

b. The clearance order tags issued by the control room must be hung by an operator and an electrician.

c. The switching order issued by Doniphan must be hung by a lineman and an electrician.

d. The clearance order tags issued by the control room must be hung by a Humboldt lineman and an operator.

ANSWER: RO 69

a. A switching order must be hung by a qualified switchman and the clearance order tags must be hung by an operator.

Distracters b., c., and d: Both a switching order and a clearance order are required. An electrician is not required.

DISTRACTOR VALIDITY:
Some components in the 12.5 distribution system require switching orders and some must be operated by an electrician, qualified switchman, lineman, or an operator.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	24786	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3		NEW

Topic Area	Description
None	HOLD 2009 NRC

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115C0C010A	Discuss the following as described in Administrative Procedure 9.EN-RP-207, Planned Special Exposure: Precautions and limitations

Related References
10CFR55.43(b)12

Related Skills (K/A)	
2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10) (3.5/3.6)

QUESTION: 70 24786 (1 point(s))

During the current calendar year a female Station Operator had received a Planned Special Exposure (PSE) of 4600 mRem (TEDE) and an Occupational Exposure of 350 mRem (TEDE).

- The female Station Operator has just filled out the required paperwork and is now a Declared Pregnant worker and has received no dose since conception.
- A job specific RWP has been created for special checks that the Reactor Building station operator must do that shows dose rates of 0.5R/hr in the area.

What is the longest time this station operator be in the area for these checks without receiving any special approvals?

a. 30 minutes

b. 45 minutes

c. 55 minutes

d. 1 hour 10 minutes

ANSWER:

c. 55 minutes

Per 9.EN-RP-207, 3.3 LIMITS FOR PSES

3.3.3 Individuals may receive dose from PSEs, in addition to the annual limit s according to the following :

	Annually	Lifetime
3.3 .3 .1 Total Effective Dose Equivalent.	5 rem	25 rem
3.3.3 .2 Individual organ or tissue .	50 rem	250 rem
3.3.3 .3 Eye.	15 rem	75 rem
3.3.3 .4 Skin or any extremity .	50 rem	250 rem

Per 9.ALARA.1, "Authorization to exceed 1,000 mrem on-site requires written approval of the individual's Department Supervisor, the ALARA Supervisor, and shall be documented on the CNS RP-9."

Per 9.ALARA.9 Per 10CFR20.1208, the Licensee shall ensure the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0 .5 rem .

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	10837	01	3/28/2005	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	3	Multiple Choice	BANK

Topic Area	Description
Systems	INT0320115E0E0500

Related Lessons	
COR0012102	Refueling

Related Objectives	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related References
9.AIARA.4 10CFR55.4112

Related Skills (K/A)	
2.3.12	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10) (3.2/3.7)

QUESTION: RO 71

For which one of the following areas must a Special Work Permit (SWP) be generated before an operator can enter it for station tours?

a. Valve room in which the highest general radiation level is 125 mem./hr.

b. An area in which loose contamination measure 40,000 dpm per 100 cm2.

c. A room in which unknown airborne radioactive material activity is 1E-9 μCi/ml.

d. An area in which fixed contamination exceeds 20 dpm/100 cm2 from an alpha emitter.

ANSWER: RO 71

a. Valve room in which the highest general radiation level is 125 mem./hr.

Explanation:

Valve room in which the highest general radiation level is 125 mem./hr. Procedure 9.ALARA.4, step 8.1.2 stated that an SWP is required for a high radiation area, highly contaminated area, or airborne radioactivity area. The only area, per 9.RADOP.2 from the above selections that fits one of these 3 categories is selection A since a High Radiation Area is defined as an area in which personal dose could exceed 0.1 rem (100 mem.) in one hour.

B, C, and D are incorrect. The limits in these distractors are all below the respective thresholds for contamination or airborne RA areas.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 72	5250	02	05/23/2009	09/18/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080618, Basis for HCTL Graph

Related Lessons	
INT0080613	OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS
INT0080608	OPS EOP Flowchart 6B - Emergency Depressurization Reactor Power (Failure-to-Scram)

Related Objectives	
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints
INT00806180010100	Using the graphs provided in the EOP and SAG Graphs Flowchart, explain how the shape of each curve or family of curves was determined.
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.
INT00806080010600	Given plant conditions and EOP flowchart 6B, EMERGENCY DEPRESSURIZATION FAILURE-TO-SCRAM and REACTOR POWER FAILURE-TO-SCRAM, state the reasons for the actions contained in the steps

Related References
NONE 10CFR55.41(b)10

Related Skills (K/A)	
2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. (CFR: 41.7 / 41.10 / 43.5 / 45.12) (3.6/4.4)

QUESTION: RO 72

The following conditions exist during a major plant transient:

. Drywell temperature	195° F
. Average Torus water temperature	200° F
. Reference leg temperature	200° F
. Primary Containment water level	12 feet
. Reactor pressure	1000 psig
. Reactor water level	+25 inches

What is expected to occur if the Automatic Depressurization System (ADS) initiated?

a. The Primary Containment Pressure Limit (PCPL) will be exceeded.

b. The Torus-to-Drywell Vacuum Breaker capacity will be exceeded.

c. The Safety Relief Valve (SRV) tail pipe supports will fail.

d. Reactor water level indication will be lost.

ANSWER: RO 72

a. The Primary Containment Pressure Limit (PCPL) will be exceeded.

Current conditions place the plant on the unsafe side of the HCTL Curve. An ADS initiation now will result in exceeding the PCPL due to insufficient energy absorption capacity to handle a blowdown.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 73	9661	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
INT0320103 CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives
INT032010300E010A Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Precautions and limitations
INT032010300E010C Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Annunciator identification

Related References
10CFR55.41(b)10

Related Skills (K/A)
2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12) (4.2/4.2)

QUESTION: RO 73

The plant is operating at near rated power when the following indications occur:

Reactor power rises to 103%
Reactor Pressure rises to 1028 psig
Reactor water level lowers to 30 inches NR
MSL ‘A’ inboard isolation valve is intermediate

Which of the following will be alarming?

a. 9-5-2/G-1 Reactor Water Low Level

b. 9-5-2/F-1 Reactor High Pressure

c. 9-5-1/A-7 APRM RPS CH A Upscale Trip or INOP

d. 9-5-1/D-4 RBM Upscale/INOP

ANSWER: RO 73

b. 9-5-2/F-1 Reactor High Pressure

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 74	20493	0	03/18/2004	09/18/2008	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	5	Multiple Choice	BANK

Topic Area	Description
Administrative	SKL0140203, Designation & location for Rx Mode Switch

Related Lessons
SKL0140203 Electrical Drawings for Licensed Operators

Related Objectives	
SKL0140203001070C	Given a Component Tabulation from a General Electric Diagram, determine the following: switch designation and location for applicable control switches

Related References	
791E256	Reactor Protection System Elementary Electrical

Related Skills (K/A)	
2.2.41	Ability to obtain and interpret station electrical and mechanical drawings. (CFR: 41.10 / 45.12 / 45.13) (3.5/3.9)

QUESTION: RO 74

The Main Steam Line Isolation Valve Closure Scram Bypass is accomplished by energizing relay 5A-K11A.

Which RPS switch is used to energize relay 5A-K11A <u>and</u> which panel is this switch located on?

a. 5A-S1, panel 9-5

b. 5A-S2A, panel 9-17

c. 5A-S4, panel 9-5

d. 5A-S6, panel, 9-16

ANSWER: RO 74

a. 5A-S1, panel 9-5

Explanation: Relay 5A-K11A is energized by the Rx Mode Switch 5A-S1 located on panel 9-5 as shown of 791E256 sheet 9.

b. 5A-S2A is a scram test switch
c. 5A-S4, is the discharge volume high water level bypass switch
d. 5A-S6, is a rod scram test switch

GIVE STUDENT GE PRINT 791E256.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
RO 75	24792	0	NEW	NEW	NA	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3		NEW

Topic Area	Description
None	HOLD 2009 NRC

Related Lessons
COR0021803 Reactor Core Isolation Cooling

Related Objectives
COR0021802001120A Given plant conditions, determine if the following RCIC actions should occur: RCIC system initiation

Related References
10CFR55.41(b)10

Related Skills (K/A)
2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13) (3.8/4.0)

QUESTION: RO 75

The plant was operating at near rated power when the following occurred:

A fire has occurred in RHR HX Room A
RCIC ISOLATION SWITCH has been placed to ISOL (isolate) at Panel 9-30

What RCIC automatic functions are bypassed due to being in ISOL (isolate)?

a. PCIS Group 5 Isolation.

b. RCIC Turbine trip on RCIC Turbine Exhaust High Back Pressure.

c. All RCIC auto initiations.

d. All RCIC Turbine Trips.

ANSWER: RO 75

c. All RCIC auto initiations.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 1	9677	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	L	1	3	Matching	NEW

Topic Area	Description
Emergency Operating Procedures	HOLD NRC 2009

Related Lessons
INT0320131 CNS Abnormal Procedures (RO) Electrical

Related Objectives
INT0320131S0S0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)
295003.AA2.03 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13) Battery status: Plant-Specific (3.2/3.5)

QUESTION: SRO 1

The plant was operating at near rated power when the following occurred:

Loss of all off site power
Both EDGs failed to start and can not be started

If this can not be corrected what is the effect on station DC loads and what procedure is required to be entered to mitigate this effect?

a. DC loads will be available for 8 hours; 5.3SBO

b. DC loads will be available for 8 hours; 5.3EMPWR

c. DC loads will be available for 4 hours; 5.3SBO

d. DC loads will be available for 4 hours; 5.3EMPWR

ANSWER: SRO 1

c. DC loads will be available for 4 hours; 5.3SBO

The USAR states DC batteries for 125 vdc and 250 vdc are design be available for 4 hours following a station blackout. CNS batteries are 8 hour batteries but credit is only given for the 4 hours. 5.3SBO has action to shed DC loads to extend battery life. This guidance is not in 5.3EMPWR.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 2	24747	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Technical Specifications, ODAM, TRM	

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Related References	
B3.6.2.2	Suppression Pool Water Level
10CFR55.43(b)5	

Related Skills (K/A)	
295030.EA2.01	Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Suppression pool level (4.1*/4.2*)

QUESTION: SRO 2

Using the information from PMIS on the following page which EOP is required to be entered and what action is required:

a. EOP-1A RPV Control; Continue injection form all available sources

b. EOP-2A Emergency RPV Depressurization; Emergency Depressurize

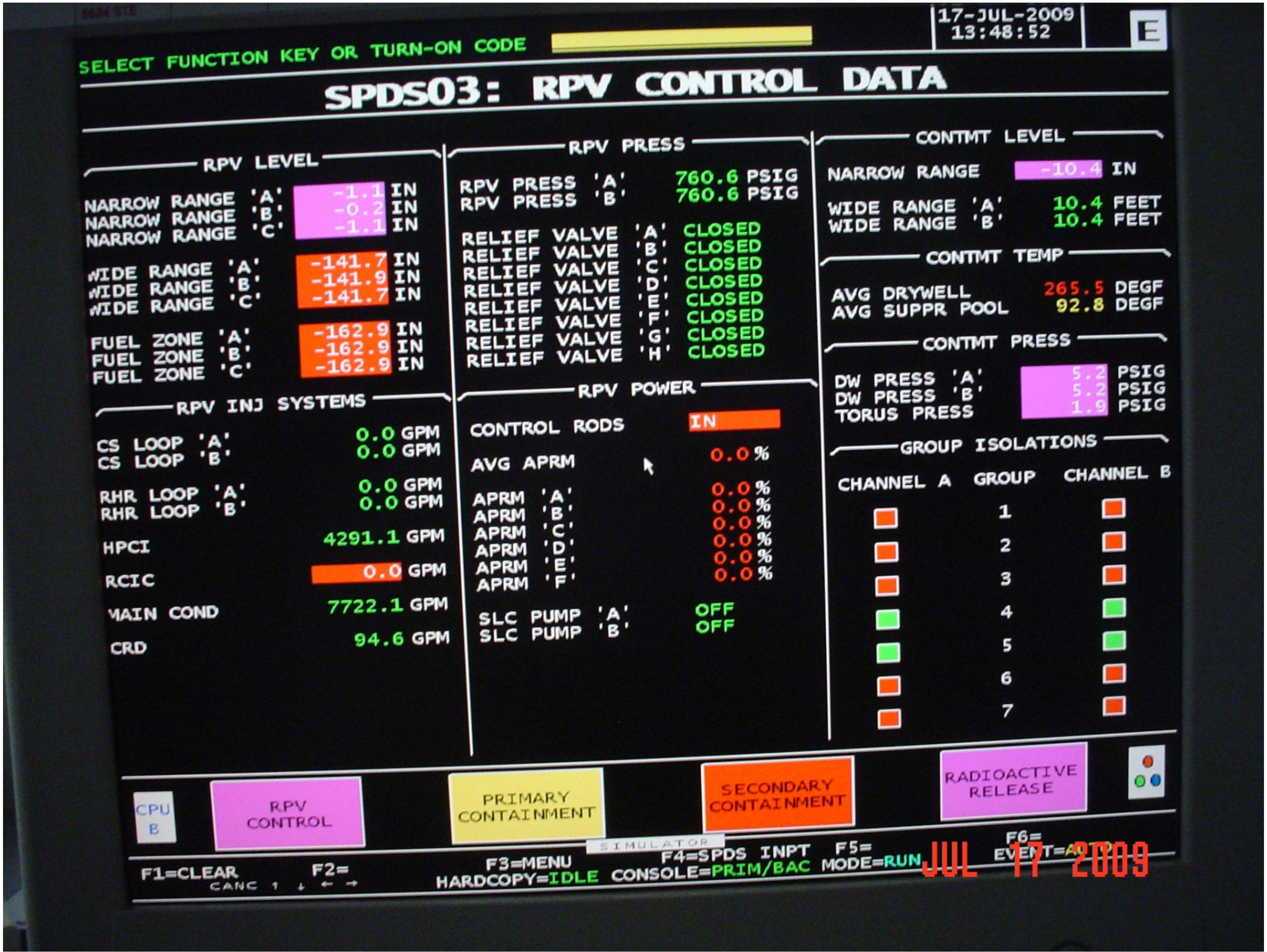
c. EOP-3A Primary Containment Control; Place HPCI in Pull-to-lock

d. EOP-3A Primary Containment Control; Place both loops of RHR in suppression Pool Cooling

ANSWER: SRO 2

e. c. EOP-3A Primary Containment Control; Place HPCI in Pull-to-lock

Even though water level is below TAF it is above -183 FZ corrected and EOP-1A would direct continuing injection from all sources. With torus level below 11 ft HPCI must be placed in PTL. The given information would not require entry into EOP-2A for Emergency Depressurization at this point.


SELECT FUNCTION KEY OR TURN-ON CODE
17-JUL-2009
13:48:52
E
SPDS03: RPV CONTROL DATA
RPV LEVEL
NARROW RANGE 'A' -1.1 IN
NARROW RANGE 'B' -0.2 IN
NARROW RANGE 'C' -1.1 IN
WIDE RANGE 'A' -141.7 IN
WIDE RANGE 'B' -141.9 IN
WIDE RANGE 'C' -141.7 IN
FUEL ZONE 'A' -162.9 IN
FUEL ZONE 'B' -162.9 IN
FUEL ZONE 'C' -162.9 IN
RPV INJ SYSTEMS
CS LOOP 'A' 0.0 GPM
CS LOOP 'B' 0.0 GPM
RHR LOOP 'A' 0.0 GPM
RHR LOOP 'B' 0.0 GPM
HPCI 4291.1 GPM
RCIC 0.0 GPM
MAIN COND 7722.1 GPM
CRD 94.6 GPM
RPV PRESS
RPV PRESS 'A' 760.6 PSIG
RPV PRESS 'B' 760.6 PSIG
RELIEF VALVE 'A' CLOSED
RELIEF VALVE 'B' CLOSED
RELIEF VALVE 'C' CLOSED
RELIEF VALVE 'D' CLOSED
RELIEF VALVE 'E' CLOSED
RELIEF VALVE 'F' CLOSED
RELIEF VALVE 'G' CLOSED
RELIEF VALVE 'H' CLOSED
RPV POWER
CONTROL RODS IN
AVG APRM 0.0 %
APRM 'A' 0.0 %
APRM 'B' 0.0 %
APRM 'C' 0.0 %
APRM 'D' 0.0 %
APRM 'E' 0.0 %
APRM 'F' 0.0 %
SLC PUMP 'A' OFF
SLC PUMP 'B' OFF
CONTMT LEVEL
NARROW RANGE -10.4 IN
WIDE RANGE 'A' 10.4 FEET
WIDE RANGE 'B' 10.4 FEET
CONTMT TEMP
AVG DRYWELL 265.5 DEGF
AVG SUPPR POOL 92.8 DEGF
CONTMT PRESS
DW PRESS 'A' 5.2 PSIG
DW PRESS 'B' 5.2 PSIG
TORUS PRESS 1.9 PSIG
GROUP ISOLATIONS
CHANNEL A GROUP CHANNEL B
1
2
3
4
5
6
7
CPU B
RPV CONTROL
PRIMARY CONTAINMENT
SECONDARY CONTAINMENT
RADIOACTIVE RELEASE
SIMULATOR
F1=CLEAR F2= F3=MENU F4=SPDS INPT F5= F6=
CANC
HARDCOPY=IDLE CONSOLE=PRIM/BAC MODE=RUN
JUL 17 2009

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 3	24751	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
INT0320199 CNS Manual Scram Action

Related Objectives
INT0320122I0I0100 Given plant condition(s), determine from memory if a reactor scram or an emergency shutdown from power is required due to the event(s).

Related References
10CFR55.43(b)2

Related Skills (K/A)
295005.AA2.04 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : (CFR: 41.10 / 43.5 / 45.13) Reactor pressure (3.7/3.8)

QUESTION: SRO 3

The plant is operating at near rated power when the following occurred:

- Main Generator MVARs begin to steadily increase
- Doniphan contacts you to inform you the grid is becoming unstable
- PCB 3310 trips open followed by PCB 3312

What effect does this have on the Reactor and what procedure must be entered to mitigate this effect?

a. Reactor Level will lower rapidly; 2.4RXLVL must be entered.

b. Reactor Power will rise rapidly; 2.4RXPWR must be entered.

c. Reactor Pressure will lower rapidly; 2.4DEH must be entered.

d. Reactor Pressure will rise rapidly; 2.1.5 Reactor Scram must be entered.

ANSWER: SRO 3

d. Reactor Pressure will rise rapidly; 2.1.5 Reactor Scram must be entered.

This is a load reject with bypass reactor pressure will rise rapidly and a scram signal will result based on Emergency trip header setpoint. 2.1.5 must be entered on any automatic reactor scram.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 4	24749	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	3	Multiple Choice	NEW

Topic Area	Description
Emergency Operating Procedures	HOLD NRC 2009

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives
INT00705070010800 From memory, state the actions required in less than or equal to one hour if two SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment, or during OPDRVs (LCO 3.6.4.3).

Related References
10CFR55.43(b)2

Related Skills (K/A)
295023.2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12) (3.6/4.6)

QUESTION: SRO 4

The plant is in a refueling outage moving recently irradiated fuel from the reactor to the SFP when a Group 6 Isolation occurs:

- Standby Gas Treatment (SBGT) A can not be started
- Standby Gas Treatment B was required to be manually started

What actions are required by Technical Specifications?

a. Enter LCO 3.0.3 immediately.

b. Restore SBGT A to OPERABLE within 7 days.

c. Verify SBGT B in Operation and OPERABLE immediately.

d. Suspend movement of recently irradiated fuel assemblies in secondary containment immediately.

ANSWER:

d. Suspend movement of recently irradiated fuel assemblies in secondary containment immediately.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 5	24750	00	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	5	Multiple Choice	NEW

Topic Area	Description
Technical Specifications	Loss of SDC mode 4

Related Lessons
INT0070505 CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050011100	From memory, in MODE 4, state the actions required in less than or equal to one hour for one or two RHR shutdown cooling subsystems inoperable (LCO 3.4.8).

Related References	
2.4SDC	Shutdown Cooling Abnormal
10CFR55.43(b)2	

Related Skills (K/A)	
295021.2.2.37	Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12) (3.6/4.6)

QUESTION: SRO 5

With the plant shutdown in Mode 4 and RHR loop A operating in Shutdown Cooling, the following conditions exist:

- Reactor pressure is 0 psig
- Recirc suction temperature is 180°F
- Reactor water level is 36" (NR)

At 0800 RPS 'B' is lost and cannot be restored for 12 hours.

What action is required first by Technical Specifications?

a. Initiate actions to suspend OPDRVs immediately.

b. Fully insert all insertable control rods by 0900.

c. Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem by 0900.

d. Restore required SRMs to OPERABLE status by 1200.

ANSWER: SRO 5

c. Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem by 0900.

Explanation:

Loss of RPS 'B' shuts RHR-MO-17 and RHR-MO-18 thus preventing RHR SDC flowpath. TS 3.4.8 addresses actions required for these conditions.

Provide student TS 3.3.1.2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 6	5221	01	08/11/2003	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	4	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070509, CNS Tech. Spec. 3.8, Electrical Power System

Related Lessons
INT0070509 OPS Tech. Spec. 3.8, Electrical Power Systems

Related Objectives	
INT00705090010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.
INT00705090010100	Given a set of plant conditions, recognize non-compliance with a Section 3.8 LCO.

Related References	
3.5.1	ECCS Operating
3.8.4	DC sources - Operating
10CFR55.43(b)2	

Related Skills (K/A)	
264000.K1.01	Knowledge of the physical connections and/or cause- effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) A.C. electrical distribution (3.8/4.1)

QUESTION: SRO 6

Given the following conditions:

- Reactor power is 32%
- 0800 9/18 BKR 1FS AUTO CLOSURE NOT PERMITTED alarm is received.
- 0900 9/18 The TB Station Operator report 125 VDC Panel DG-1 has no power.

If this equipment CANNOT be returned to operable condition, when is the reactor required to be in MODE 3?

a. 2200 on 9/18.

b. 2300 on 9/18.

c. 2100 on 9/19.

d. 2200 on 9/19.

ANSWER: SRO 6

c. 2100 on 9/19.

EXPLANATION OF ANSWER: The loss of the 1FS closure capability required one offsite source to be declare INOP this gives 7 days. With addition of 1 EDG this reduces that time to 24 hours to restore then 12 hours to be in mode 3. This would be 36 hours from 0900 on 9/18.

REFERENCES: Tech Spec 3.8.1, 3.8.4

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 7	24752	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	NEW

Topic Area	Description
Emergency Operating Procedures	INT0080606, Alternate Power indications & required action

Related Lessons
INT0080606 FLOWCHART 6A - RPV PRESSURE/POWER (FAILURE-TO-SCRAM) INT0080605 OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806060011100	Given plant conditions and the EOP Flowchart 6A, RPV PRESSURE/POWER FAILURE-TO-SCRAM, determine required actions.
INT00806050010300	Identify indications of reactor power level other than APRMs to determine reactor power is 3.0%.

Related References
10CFR55.43(b)5

Related Skills (K/A)
295006.AA2.01 Ability to determine and/or interpret the following as they apply to SCRAM : (CFR: 41.10 / 43.5 / 45.13) Reactor Power (4.5*/4.6*)

QUESTION: SRO 7

The plant was at near rated power when a DEH failure occurred:

- Reactor Pressure maximum was 1060 psig
- The Main turbine tripped
- All 8 white RPS lights are OFF on panel 9-5
- Currently all 3 Main Turbine bypass valves are fully open
- 2 SRVs are full open and a 3rd SRV is cycling

What is power and what procedure is required to be entered first?

a. Reactor power is 10%-30%; Enter EOP-1A

b. Reactor power is 10%-30%; Enter EOP-6A

c. Reactor power is 40%-60%; Enter EOP-1A

d. Reactor Power is 40%-60%; Enter EOP-6A

ANSWER: SRO 7

c. Reactor power is 40%-60%; Enter EOP-1A

Explanation:

With the BPV open and 2 fully open SRV with an additional SRV cycling this steam flow correlates to approximately 50% Reactor power. EOP-6A can not be entered directly it must be entered from EOP-1A

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 8	24753	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
INT0320128 CNS Abnormal Procedures (RO) Containment

Related Objectives
INT0320128I0I0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)
295010. AA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE : (CFR: 41.10 / 43.5 / 45.13) Drywell pressure (3.8/3.9)

QUESTION: SRO 8

The plant is operating at near rated power when the following occurs:

- The Reactor Operator identifies an increasing trend in Drywell Pressure,
- DW Pressure is currently 0.9 psig rising slowly.
- There is a corresponding rising trend in Drywell Temperature from 105F to 140F.
- The BOP noted that all three channels of the Drywell Radiation Monitor are operating and the trend shows NO change in activity.

Which of the following problems would give this indication and what procedure should be entered?

a. Failure of Drywell fan coil units; 2.4PC Primary Containment Control

b. Small water leak in the Drywell; 2.1.5 Reactor Scram

c. Small Steam Leak in the Drywell; EOP 3A Primary Containment Control

d. Nitrogen leak in the Drywell; 2.4PC Primary Containment control

ANSWER: SRO 8

a. Failure of Drywell fan coil units; 2.4PC Primary Containment Control

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 9	113	0	09/10/1998	09/18/2008	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	1	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070504, CNS Tech Spec 3.3, Instrumentation

Related Lessons
INT0070504 CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705040010100	Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.

Related References	
3.3.4.1	Anticipated transient without scram recirculation pump trip (ATWS-RPT) instrumentation
10CFR55.43(b)2	

Related Skills (K/A)
295007.2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3) (3.4/4.7)

QUESTION: SRO 9

The plant is operating at 85% power on 5/10. At 1035, it is discovered that one (1) of the channels (NBI-PS-102C) for high reactor pressure that initiates ATWS-RPT will not trip. All other instrumentation associated with ATWS-RPT function normally.

What actions are required by Technical Specifications?

a. Remove the associated recirculation pump from service OR be in MODE 2 within 6 hours.

b. Restore the channel to operable status OR place the inoperable channel in the tripped condition by 1135 on 5/10.

c. Restore the channel to operable status OR place the inoperable channel in the tripped condition by 1035 on 5/13.

d. Restore the channel to operable status OR place the inoperable channel in the tripped condition by 1035 on 5/24.

ANSWER: SRO 9

d. Restore the channel to operable status OR place the inoperable channel in the tripped condition by 1035 on 5/24.

Provide REFERENCE: TS 3.3.4.1

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 10	24754	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	2	Multiple Choice	NEW

Topic Area	Description

Related Lessons
INT0320122 CNS Abnormal Procedures (RO) CRD Control Rods

Related Objectives
INT0320122G0G0100 Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)
295022.2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.8/4.0)

QUESTION: SRO 10

The plant is in Start up at 500 psig for the Drywell walkdowns

- CRD pump A is tagged out for maintenance to finish work
- CRD pump B trips and can not be restarted

What procedure is required to be entered and action should be taken to mitigate these conditions?

a. Enter 2.4CRD CRD Trouble which requires operators to stop all rod moves until a CRD pump can be restarted.

b. Enter ARP 9-5-2/A-6 CRD Pump Breaker Trip which requires operators to scram the Reactor.

c. Enter 2.4CRD CRD Trouble which requires operators to scram the Reactor.

d. Enter ARP 9-5-2/A-6 CRD Pump Breaker Trip which requires operators to take manual control of CRD Flow Control Valves and adjust to 50 gpm.

ANSWER: SRO 10

b. Enter ARP 9-5-2/A-6 CRD Pump Breaker Trip which requires operators to scram the Reactor.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 11	24757	00	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons	
INT0320135	CNS Abnormal Procedures (RO) - Condensate/Feedwater

Related Objectives	
INT0320135G0G0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)	
259002.A2.02	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: (CFR: 41.5 / 45.6) Loss of any number of reactor feedwater flow inputs (3.3/3.4)

QUESTION: SRO 11

The plant is operating at near rated power with RFC-FT-50B bypassed due to erratic behavior last shift causing the RVLC system to shift to single element. When the following occurs:

- RFC-FT-50A fails downscale

How does this affect the plant and what action is required?

a. Both RFPT-1A and 1B transfer to MDVP. Enter 2.4RXLVL, RPV Water Level Control Trouble and control Reactor water level manually.

b. Both RFPT-1A and 1B transfer to MDEM. Enter 2.4RXLVL, RPV Water Level Control Trouble and control Reactor water level manually.

c. Both Reactor Recirc Pumps run back to 22%. Enter 2.4RR, Reactor Recirculation Abnormal and take actions for operation in the Stability Exclusion area.

d. Both Reactor Recirc Pumps run back to 45% and stop. Enter 2.4RR, Reactor Recirculation Abnormal and take actions for Reactor Recirc Runback.

Answer: SRO 11

c. Both Reactor Recirc Pumps run back to 22% and stop. Enter 2.4RR, Reactor Recirculation Abnormal and take actions for operation in the Stability Exclusion area.

Explanation:

With RFC-FT-50B bypassed due to erratic behavior last shift. The loss of 50A will send a signal to the Reactor Recirc Pumps to run back to minimum (22% setpoint) because it senses a loss of feedwater. With power starting at near 100% this will drive the plant deep into the stability exclusion area. This is entry into 2.4RR.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 12	24758	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Integrated Plant	HOLD NRC 2009

Related Lessons
INT0320131 CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131S0S0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)	
264000.A2.01	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: (CFR: 41.5 / 45.6) Parallel operation of emergency generator (3.5/3.6)

QUESTION: SRO 12

The plant is operating at near rated power when the following occur

- 6.1DG.101 Diesel Generator 31 day Operability Test is ongoing with 2 hours of the run completed on EDG A
- A fault results in the trip and lockout of 4160V Bus 1A Breaker 1AN

What are the consequences of this and what actions must be taken?

a. EDG A will be carrying both 4160F and 4160 A; Rapidly Reduce power per 2.1.10 to prevent overloading EDG A

b. EDG A will over speed and trip resulting in the loss of 4160F and 4160A due to this loss the Reactor will scram due to loss of RPS power; Enter 2.1.5 Reactor Scram and perform mitigating actions

c. Breaker 1AF will trip resulting in the loss 4160A, EDG A will trip and the ESST will energize 4160F; Enter 5.3Grid and Manually Scram the Reactor.

d. Breaker 1AF will trip resulting in the loss 4160A, EDG A will continue to power 4160F; Enter ARP C-2/A1 4160V Bus 1A Undervoltage and Scram the Reactor.

ANSWER: SRO 12

d. Breaker 1AF will trip resulting in the loss 4160A, EDG A will continue to power 4160F; Enter ARP C-2/A1 4160V Bus 1A Undervoltage and Scram the Reactor.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 13	5297	01	07/21/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Technical Requirements Manual	INT0070605, Does LOSF exist & when to MODE 3? HPCI Min Flow, RCIC Tripped (No LOSF)

Related Lessons	
INT0070605	OPS TRM - Safety Function Determination Program and Other TRM Required Programs
INT0070504	CNS Tech. Spec. 3.3, Instrumentation
INT0070506	OPS Tech. Spec. 3.5, Emergency Core Cooling (ECCS) and Reactor Core Isolation Cooling (RCIC) System

Related Objectives	
INT00706050010500	For the Safety Function Determination Program (SFDP), calculate the Maximum Completion TIME (MCT) for supported system inoperabilities.
INT00705040010100	Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.
INT00706050010300	Given plant conditions, TS and the TRM, determine if a Safety Function Determination (SDF) is required.
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705060010100	Given a set of plant conditions, recognize non-compliance with a Section 3.5 LCO.
INT00705060010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.5 LCO, determine the ACTIONS that are required.

Related References	
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation
3.5.3	RCIC system
3.0.6	LCO Applicability

T5.11	Safety Function Determination Program
5.5.11	Safety Function Determination Program (SFDP)
10CFR55.43(b)2	

Related Skills (K/A)	
217000.2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12) (4.0/4.6)

QUESTION: SRO 13

The plant is operating at rated power with the following conditions:

- 6/3 at 0300, HPCI flow switch wires that feed into the minimum flow valve circuit are broken off by a trainee on a tour.
- 6/3 at 0800, it is discovered that RCIC is tripped without causing an alarm. The trip cannot be reset.

If NO equipment is repaired or returned to service, does a Loss of Safety Function (LOSF) exist (as defined by Technical Specifications) AND what is the EARLIEST date and time Technical Specifications requires the unit to be in MODE 3?

a. No. 6/3 at 2000

b. No. 6/10 at 1500

c. Yes. 6/3 at 2100

d. Yes. 6/10 at 1600

ANSWER: SRO 13

a. No. 6/3 at 2000

NO LOSF exists as RCIC is not assumed by any accident analysis and ADS remains available.

6/3 at 0300, enter 3.3.5.1.A & E. The function lost is 3.f, so E.1 does not apply. 7 days to call HPCI inop (0300 on 6/10), then 14 days to restore HPCI to operable status (0300 on 6/24) then MODE 3 in 12 hours (1500 on 6/24)

6/3 at 0800, 3.5.3.A and 3.5.3.B is entered (be in MODE 3 within 12 hours) = 6/3 at 2000. You DO NOT get the hour as the 1 hour is to verify operability. With HPCI known to be inoperable, must enter 3.5.3.B immediately.

GIVE STUDENT TS 3.5.1 and 3.5.3

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 14	4064	03	05/05/1994	09/18/2008	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Administrative	COR0099900 QUESTION #329

Related Lessons	
COR0099900	NRC Licensed Personnel Requalification Written Exams
COR0023002	SOURCE RANGE MONITOR
INT0320104	CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
COR00999000000100	Open Reference Question
COR0023002001060A	Describe the SRM system design features and/or interlocks that provide for the following: Rod withdrawal blocks
COR0023002001100A	Given plant conditions, determine if any of the following SRM related actions should occur: Control rod withdrawal block.
INT032010400A0100	Discuss Precautions and Limitations outlined in General Operating Procedure 2.1.1, Startup Procedure.

Related References	
2.1.1	Startup Procedure
2.1.1.2	Technical Specifications Pre-Startup Checks
10CFR55.43(b)2	

Related Skills (K/A)	
215004.2.2.37	Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12) (3.6/4.6)

QUESTION: SRO 14

A reactor startup is being initiated and the RO is ready to withdraw the first control rod. The RO reports that SRMs B and C are reading >3 cps, SRMs A and D are reading 2 cps.

Which of the following is correct regarding the startup?

a. Withdraw control rods since at least one SRM is reading > 3 cps and rod block is now cleared.

b. Withdraw control rods since only two SRMs are required to be fully inserted and reading > 3 cps for a startup.

c. Stop since at least 3 SRMs must be fully inserted and reading > 3 cps to clear the rod block.

d. Stop since Tech Specs allows only one SRM to read < 3 cps for a startup.

ANSWER: SRO 14

c. Stop since at least 3 SRMs must be fully inserted and reading > 3 cps to clear the rod block.

Explanation:
T.S. allows 2 downscale; however only 1 may be manually bypassed to clear downscale Rod Block.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 15	20182	00	02/03/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
2	H	1	4	Multiple Choice	BANK

Topic Area	Description
Abnormal and Emergency Procedures	2.4CSCS HPCI

Related Lessons
INT0320136 OPS-CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT0320136L0L0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References
10CFR55.43(b)5

Related Skills (K/A)	
206000.A2.11	Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Low reactor water level: BWR-2,3,(4.1/4.2)

QUESTION: SRO 15

The plant is operating at near rated conditions when HPCI initiation occurs. The plant indications are as follows:

- Reactor Pressure 930 psig
- Reactor water level 35” and steady
- Drywell pressure 0.34 psig
- Drywell temperature 134°F

What effect does this have on the plant and what actions must be taken to mitigate this condition?

a. HPCI will inject resulting in Reactor Vessel Level Control (RVLC) reducing Reactor Feed Pump speed to maintain reactor water level in the normal band. Enter 2.4RXLVL and monitor RVLC for proper operation.

b. HPCI will inject until Reactor water level reaches 50 inches and trips. Enter 2.4RXLVL and control Reactor water level manually.

c. HPCI will inject resulting in a Scram. Enter 2.4CSCS and place HPCI in PTL to prevent injection.

d. HPCI will inject resulting in a Scram. Enter 2.1.5 Reactor Scram and perform the mitigating actions.

ANSWER: SRO 15

c. HPCI will inject resulting in a Scram. Enter 2.4CSCS and place HPCI in PTL to prevent injection.

Explanation:

The initiation of HPCI with no valid signal is and entry condition to 2.4CSCS and the actions are to place HPCI in PTL to prevent HPCI injection which will result in a high APRM flux scram.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 16	24761	0	NEW	NEW	NA	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	NEW

Topic Area	Description
Emergency Operating Procedures	HOLD NRC 2009

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives
INT00705070010300 Given a set of plant conditions that constitutes non-compliance with a Chapter 3.6 LCO, determine the ACTIONS that are required.

Related References
10CFR55.43(b)1 10CFR55.43(b)2

Related Skills (K/A)
290002.A2.02 Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) †Overpressurization transient (3.6/3.9)

QUESTION: SRO 16

The plant is operating at 65% power while performing 6.MS.201 Main Steam Isolation Valve Operability Test (IST).

During conduct of the test the following closure times where noted:

MS-AO-80A	3.5 seconds
MS-AO-86A	3.5 seconds
MS-AO-80B	4 seconds
MS-AO-86B	2.5 seconds
MS-AO-80C	2 seconds
MS-AO-86C	4 seconds
MS-AO-80D	4.5 seconds
MS-AO-86D	1.5 seconds

What is the effect on the plant and what actions are required (if any)?

a. Due to stroke times less than 3 seconds the plant is outside its design analysis for the MSIV closure transient; with all stroke times less than 5 seconds there is no required entry into TS.

b. Due to stroke times greater than 3 seconds the plant is outside its design analysis for the MSIV closure transient; Enter TS 3.6.1.3 Condition A which requires isolating MSL-A, MSL-C, and MSL-D within 8 hours.

c. Due to stroke times greater than 3 seconds the plant is outside its design analysis for the MSIV closure transient; Enter TS 3.6.1.3 Conditions A and B which requires isolating MSL-A within 1 hour.

d. Due to stroke times less than 3 seconds the plant is outside its design analysis for the MSIV closure transient; Enter TS 3.6.1.3 Condition A which requires isolating MSL-B, MSL-C, and MSL-D within 8 hours.

ANSWER: SRO 16

d. Due to stroke times less than 3 seconds the plant is outside its design analysis for the MSIV closure transient; Enter TS 3.6.1.3 Condition A which requires isolating MSL-B, MSL-C, and MSL-D within 8 hours.

Provide student with TS 3.6.1.3.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 17	23388	00	07/11/2006	09/18/2008	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070505, what has to be done if one jet pump is inop and why. (ILT 2006 AUDIT EXAM)

Related Lessons
INT0070505 CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.
INT00705050010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.4 Specification.

Related References	
3.4.2	Jet pumps
10CFR55.43(b)2	

Related Skills (K/A)	
20002.2.2.42	Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3) (3.9/4.6)

QUESTION: SRO 17

Following a transient the Reactor Operator reports that the indicated d/p for Jet Pump #6 and Jet Pump flow has fallen outside the Operability Curves.

What is the action required and what are the bases for that action?

a. Declare Jet Pump #6 inoperable because it could increase the possibility of improper core flow indications to the APRMs.

b. Declare Recirc Loop B inoperable in accordance with LCO 3.4.1 Recirculation Loops Operating, because it could cause the core flow coast down time to increase.

c. Declare Recirc Loop B inoperable in accordance with LCO 3.4.1 Recirculation Loops Operating, because it could cause the loss of the ability to maintain 2/3 core height during a LOCA.

d. Declare Jet Pump #6 inoperable in accordance with LCO 3.4.2 Jet Pumps, because it could increase the blowdown area and reduce the capability to reflood during a designed basis LOCA.

ANSWER: SRO 17

d. Declare Jet Pump #6 inoperable in accordance with LCO 3.4.2, because it could increase the blowdown area and reduce the capability to reflood during a designed basis LOCA.

Explanation: In accordance with Tech Specs 3.4.2, Jet Pumps; SR 3.4.2.1 indicates that either Recirc pump flow to speed and loop flow to speed fall within established patterns. Or that each Jet Pump diffuser to lower plenum d/p falls within established patterns. This is verified by comparing the indications to Operability Curves maintained in the Control Room. Given that both are outside those patterns, the jet pump must be declared inoperable per T.S. 3.4.2. The reason given in bases A.1, An inoperable jet pump can increase the blowdown area and reduce the capability to reflood during a designed basis LOCA.

Provide Student TS 3.4.1 and 3.4.2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 18	16420	04	06/23/2006	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080617, EOP 5A, when to scram and why on hi Sec Cntmt water level

Related Lessons	
INT0080617	OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

Related Objectives	
INT00806170010600	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, determine required actions.
INT00806170010700	Given plant conditions and EOP flowchart 5A, SECONDARY CONTAINMENT CONTROL and RADIOACTIVITY RELEASE CONTROL, state the reasons for the actions contained in the steps.

Related References	
5.8	Emergency Operating Procedures (EOPs)
10CFR55.43(b)5	

Related Skills (K/A)	
268000.2.2.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12) (4.2/4.4)

QUESTION: SRO 18

The plant is operating at 80% power when the following occur:

- 9-3-2/G-5, SUPPR POOL NR/WR LOW LEVEL, alarmed 15 minutes ago.
- S-1/A-1, REACTOR BLDG A SUMP HI-HI LEVEL, alarmed 10 minutes ago.
- S-1/A-3, REACTOR BLDG C SUMP HI-HI LEVEL, alarms.
- The SO dispatched to investigate reports:
  - Water level in the NW quad is 5 feet sump level and is rising fast.
  - Water level in the SW quad is 1 foot sump level and is rising slowly.
- It is determined that the leak is from the Torus and is NOT isolable.
- Primary Containment (PC) level is 11.6 feet and slowly lowering.

What EOP should be entered and what action will be required if these conditions continue?

a. Enter EOP-5A and Scram when NW quad level is above 9.5 feet because equipment required for safe shutdown is threatened.

b. Enter EOP-5A and Emergency Depressurize when NW quad water level is above 9.5 feet because equipment required for safe shutdown is threatened.

c. Enter EOP-3A and Scram when PC level reaches 11 feet due to challenging Vortex Limits (Graph 4).

d. Enter EOP-3A and Emergency Depressurize PC level reaches 11 feet because equipment required for safe shutdown is threatened.

ANSWER: SRO 18

a. Enter EOP-5A and Scram when NW quad level is above 9.5 feet because equipment required for safe shutdown is threatened.

Give Students EOP-3A and EOP-5A with entry condition and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 19	20543	01	6/29/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	3	Multiple Choice	BANK

Topic Area	Description
Administrative	Reportability Accidental Exposure

Related Lessons
INT0320203 CONDUCT OF OPERATIONS PROCEDURES (SRO)

Related Objectives	
INT03202030000900	Using procedure 2.0.5, identify conditions requiring one or four hour NRC notification. (2.0.5)

Related References
2.0.5 10CFR55.43(b)5

Related Skills (K/A)	
2.1.20	Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) (4.6/4.6)

QUESTION: SRO 19

During the transfer of radioactive waste from the site an accident occurred on the plant access road that resulted in the breach of a shipping container. A release of airborne contamination occurred. An employee walking beside the access road at the time of the accident received a significant internal dose.   After the employee is back in the protected area, further evaluation reveals the employee received 7 times the federal limit (internal exposure).

What is the most restrictive report that is required?

a. 1 hour report

b. 4 hour report

c. 8 hour report

d. 24 hour report

ANSWER: SRO 19

a. 1 hour report

Events involving byproduct, source, or special nuclear material that may have caused or threatens to cause: potential for exposure to personnel in unrestricted areas;

An individual to receive a dose equal to or in excess of the limits of 10CFR20.2202(a)(1).

Provide 2.0.5 to candidate

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 20	24486	00	12/31/2008	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	2	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	TS 5.5.7 VFTP

Related Lessons
INT0070513 CNS Technical Specifications 5.0, Administrative Controls

Related Objectives	
INT00705130010200	Given a set of conditions that constitutes non-compliance with a Chapter 5.0 Requirement, determine the actions that are required.

Related References
10CFR55.43(b)2

Related Skills (K/A)	
2.2.37	Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12) (3.6/4.6)

QUESTION: SRO 20

The plant is operating at 70% power with substantial painting being performed in the cable spreading room. A surveillance error results in a Group 6 initiation signal. What requirements must be satisfied per Technical Specification section 5.0?

a. Only SGT HEPA and charcoal filters must be tested in-place for penetration, bypass, and differential pressure.

b. Both SGT HEPA and CREF HEPA charcoal filters must be tested in-place for penetration, bypass, and differential pressure.

c. Only CREF HEPA and charcoal filters must be tested in-place for penetration, bypass, and a sample of only CREF charcoal adsorber must be laboratory tested for methyl iodide removal rate.

d. Both SGT HEPA and CREF HEPA and charcoal filters must be tested in-place for penetration, bypass, and a sample of both SGT and CREF charcoal adsorber must be laboratory tested for methyl iodide removal rate.

ANSWER: SRO 20

c. Only CREF HEPA and charcoal filters must be tested in-place for penetration, bypass, and a sample of only CREF charcoal adsorber must be laboratory tested for methyl iodide removal rate.

a. is incorrect because SGT does not have any ventilation zone communicating with the cable spreading room.
b. is incorrect for the same reason as a.
d. is incorrect for the same reason as a. and b.

Provide Student with TS Section 5.5 for CREFs

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 21	4114	01	09/03/2003	09/18/2008	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	H	1	4	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	COR0011802, KAMAN Operability SRO on loss of flow.

Related Lessons	
COR0011802	OPS Radiation Monitoring
INT0070702	ODAM Specifications
INT0070602	TRM - Instrumentation

Related Objectives	
INT00707020010300	Given the ODAM Appendix D, and conditions of non-compliance with any ODAM Specification Section 3.1 thru 3.5, determine the required actions.
INT0070602001010C	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.3.3 Non-Type A, Non-Category 1 Post Accident Monitoring PAM Instrumentation.

Related References
10CFR55.43(b)2

Related Skills (K/A)
2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9) (2.9/3.1)

QUESTION: SRO 21

The plant is at rated power when the ERP process flow signal fails low.

How is the OPERABILITY of the ERP KAMAN System affected and what action (if any) is required?

a. Neither the ERP normal nor the ERP high range KAMANs is affected by the failure of the ERP process flow. No actions are required by the TRM or ODAM.

b. Only the ERP flow instrument would be considered inoperable. DLCO 3.3.2 Required Action E.1 would have to be performed within the specified time limits. No actions are required by the TRM.

c. Both the ERP normal range KAMAN and the flow instrument would be considered inoperable. DLCO 3.3.2 Required Actions E.1 and F.1 AND F.2 would have to be performed within the completion times. No actions are required by the TRM.

d. Both the ERP normal and the ERP high range KAMANs would be considered inoperable, AND the ERP flow instrument would also be INOPERABLE. DLCO 3.3.2 Required Actions E.1, F.1 AND F.2 would have to be performed within the completion times, AND TLCO 3.3.3 Required Action B.1 and B.2 would have to be completed within their specified time limits.

ANSWER: SRO 21

d. Both the ERP normal and the ERP high range KAMANs would be considered inoperable, AND the ERP flow instrument would also be INOPERABLE. DLCO 3.3.2 Required Actions E.1, F.1 AND F.2 would have to be performed within the completion times, AND TLCO 3.3.3 Required Action B.1 and B.2 would have to be completed within their specified time limits.

Provide to student TLCO 3.3.3 and DLCO 3.3.2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 22	23182	00	08/20/2006	09/18/2008	Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Orgin
3	H	1	5	Multiple Choice	BANK

Topic Area	Description
Emergency Operating Procedures	INT0080618, Apply EOP Graphs to determine the required procedures to execute. (2006 BIENNIAL EXAM)

Related Lessons
INT0080618 OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives
INT00806180020400 Using the Cautions provided in the EOP and SAG Flowcharts, explain the bases behind each of the Cautions.

Related References
10CFR55.43(b)5

Related Skills (K/A)	
2.4.23	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. (CFR: 41.10 / 43.5 / 45.13) (3.4/4.4)

QUESTION: SRO 22

The plant is operating when a station blackout occurred. All control rods fully inserted on the scram. 1 hour later the following conditions are noted by the crew:

- Reactor pressure is 1000 psig and being controlled by manual SRV actuation.
- Drywell pressure is 4 psig and very slowly rising.
- Drywell temperature is 250^{o}F and slowly rising.
- Suppression Pool temperature is 175^{o}F and rising slowly.
- Primary Containment level is 12.5'and stable.
- Operator at Instrument rack reports rReactor water level at -38" (FZ Indicated) and slowly lowering.
- HPCI and RCIC have failed and cannot be made to inject.

What action is required?

a. Exit EOP-1A and Enter EOP-2B RPV Flooding.

b. Continue in EOP-1A and Anticipate Emergency RPV Depressurization.

c. Exit EOP-1A pressure leg and Enter EOP-2A to perform Steam Cooling.

d. Exit EOP-1A pressure leg and Enter EOP-2A to Emergency Depressurize.

ANSWER: SRO 22

c. Exit EOP-1A pressure leg and Enter EOP-2A to perform Steam Cooling.

Explanation:

No injection sources are available and reactor level is 0" Corrected FZ. EOP-1A directs that that EOP-1A pressure control actions be terminated (Override in RC/P-4).

Distractors:

a. is incorrect because level is determinate. While not availablel in the control room level indication is available locally.
b. is incorrect because thee actions are counter intuitive to the objectives of steam cooling.
d. is incorrect because ED is not required yet.

Provide student with EOP-1A with cautions and entry conditions removed

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 23	5101	01	03/17/2003	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Administrative	COR0012102001030C, Refueling

Related Lessons	
COR0012102	Refueling
INT0320117	CNS Administrative Procedures Volume Ten, Nuclear Performance Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
COR0012102001030C	Given a Reactor Refueling and Servicing Equipment manipulation, predict and explain the changes in the following parameters: Core reactivity level

Related References	
10.23	New Fuel Inspection, Channeling, and Control Blade Inspection
10.25	Refueling - Core Unload, Reload, and Shuffle
2.2.31	Fuel Handling - Refueling Platform
10CFR55.43(b)7	

Related Skills (K/A)
2.1.41 Knowledge of the refueling process. (CFR: 41.2 / 41.10 / 43.6 / 45.13) (2.8/3.7)

QUESTION: SRO 23

When handling NEW fuel in the Fuel Pool Area, how many bundles are allowed outside normal storage area OR shipping container at a time?

a. one (1)

b. two (2)

c. three (3)

d. four (4)

ANSWER: SRO 23

c. three (3)

Answer source: 10.23

SRO Justification: SRO responsible for refueling and the Special Nuclear Material Executor must be an SRO licensed individual.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 24	20513	01	07/22/2004	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
4	H	1	6	Multiple Choice	BANK

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070605, Sec. Containment inop (LOSF)

Related Lessons	
INT0070507	CNS Tech. Spec. 3.6, Containment Systems
INT0070605	OPS TRM - Safety Function Determination Program and Other TRM Required Programs

Related Objectives	
INT00705070010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.6 LCO, determine the ACTIONS that are required.
INT00706050010400	Given plant conditions, TS and TRM, determine if a Loss of Safety Function (LOSF) exists.
INT00706050010800	Given plant conditions, TS and the TRM, determine the required Technical Specification ACTION(s) if a LOSF is identified.
INT00705070010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.0.6	LCO Applicability
3.6.4.1	Secondary Containment
3.6.4.3	Standby gas treatment (SGT) system
5.5.11	Safety Function Determination Program (SFDP)
T5.11	Safety Function Determination Program
10CFR55.43(b)2	

Related Skills (K/A)	
2.2.23	Ability to track Technical Specification limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13) (3.1/4.6)

QUESTION: SRO 24

The plant is conducting a startup and is currently holding power at 10%, when the following occurs:

- 3/15 at 0100, "A" SGT could only develop -0.10 inches water gauge Reactor Building d/p.
- "A" SGT flow is 1500 SCFM and temperatures are normal.
- 3/15 at 0300, "B" SGT breaker tripped when it was started.

If NO equipment is repaired or returned to service, does a Loss of Safety Function (LOSF) exist (as defined by Technical Specifications) AND what is the EARLIEST date and time Technical Specifications requires the unit to be in MODE 3?

a. No. 3/15 at 1300

b. No. 3/22 at 1600

c. Yes. 3/15 at 1600

d. Yes. 3/15 at 1700

ANSWER: SRO 24

d. Yes. 3/15 at 1700

A LOSF exists per TS bases when Secondary Containment is no longer able to maintain adequate leak tightness to satisfy the -0.25" SR (TS bases for 3.6.4.1)

3/15 at 0100, enter 3.6.4.1.A (restore in 4 hours), expires at 0500.
3/15 at 0500, enter 3.6.4.1.B (MODE 3 in 12 hours), be in MODE 3 by 3/15 at 1700 (most limiting). 3.6.4.3 need not be entered as "A" SGT is operable, secondary containment is the problem.

3/15 at 0300, enter 3.6.4.3.A (7 day LCO), expires 3/22 at 0300, MODE 3 by 1500

Provide candidate with TS 3.6.4.1 and 3.6.4.3

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
SRO 25	8860	00	2/3/2008	09/18/2008	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Origin
3	L	1	4	Multiple Choice	BANK

Topic Area	Description
Emergency Plan	GEN0030103; Emergency Exposure Limits; SRO ONLY

Related Lessons	
GEN0030103	EMERGENCY DIRECTOR/FACILITY MANAGEMENT PERSONNEL
GEN0030401	Emergency Plan for Licensed Operators

Related Objectives	
GEN00301030000100	State who, by title, may authorize radiation exposure in excess of the occupational limits

Related References	
9.EN-RF-207	Planned Special Exposure
9.ALARA.1	Personnel Dosimetry and Occupational Radiation Exposure Program
10CFR55.43(b)4	

Related Skills (K/A)	
2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10) (3.2/3.7)

QUESTION: SRO 25

Following a LOCA, it is necessary to authorize an emergency exposure for an individual who has volunteered to enter a very high radiation area to protect valuable property.

- The individual has an accumulated TEDE of 3000 mrem for the year.
- The TSC is NOT activated.

As Emergency Director, which one of the following describes the MAXIMUM permissible dose you may authorize in accordance with EPIP Procedure 5.7.12, Emergency Radiation Exposure Control?

a. 7 REM

b. 10 REM

c. 22 REM

d. 25 REM

ANSWER: SRO 25

b. is correct. The Emergency Director may authorize 10 rem emrgency exposure to protect valuable property.

a. is incorrect. Emergency exposure limits are allowed above the 10CFR20 limits. 7 rem would be calculated if existing exposure was subtracted from the exposure limit.

c. is incorrect. Emergency exposure limits are allowed above the 10CFR20 limits. 22 rem is calculated subtracting exisiting exposure from the limit for life saving or protection of large populations.

d. is incorrect. This is the emergency exposure limit for life saving or protection of large populations.