Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000.K1.03	
	Importance Rating	2.6	

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) Lesson COR002-07-02
(Attach if not previously provided)
(including version/revision number) 29

Learning Objective: See Attached (As available)

Question Source: Bank # 15139
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	15139	01	06/02/2004		None	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	

Topic Area	Description
Electrical	COR0020702, DC Electrical Distribution

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001110B	Predict the consequences of the following events on the DC Electrical Distribution System: Loss of Battery ventilation
COR002070200 I 0200	Given conditions and/or parameters associated with the DC Electrical Distribution System, determine if related Technical Specification and Technical Requirements Manual Limiting Condition for Operation are met

Related References
COR0020702

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: 1 15139 (I point(s))

What is the main concern in the Battery Rooms during a failure of the Battery Room ventilation exhaust fans?

a. Temperatures that will result in premature battery charger failures.

b. Temperatures that will cause rapid vaporization of battery electrolyte.

c. Humidity that will result in excessive battery post corrosion and reduced current flow.

d. Temperatures and hydrogen concentration levels that will make safe operation of equipment uncertain.

ANSWER: 1 15139

d. Temperatures and hydrogen concentration levels that will make safe operation of equipment uncertain.

Explanation:

On a loss of Battery Room Ventilation removes the capability for the removal of H2 from the rooms. From the Student text - 1. The battery room exhaust fans remove the hydrogen produced by the 250 VDC and the 125 VDC batteries. 3. A loss of Battery Room Ventilation could result in excessive hydrogen levels in the room. This is a special concern when batteries are being charged, since the charging process produces hydrogen as a byproduct.

Distractors:

a. Temperatures that will result in premature battery charger failures. Although this would be a concern for the chargers the main concern according to the Student text is the accumulation of Hydrogen in the battery rooms that might get to an explosive level.

b. Temperatures that will cause rapid vaporization of battery electrolyte. The elevated temperatures will cause some electrolyte to evaporate, main concern according to the Student text is the accumulation of Hydrogen in the battery rooms that might get to an explosive level.

c. Humidity that will result in excessive battery post corrosion and reduced current flow. A concern, however the main concern according to the Student text is the accumulation of Hydrogen in the battery rooms that might get to an explosive level.

Examination Outline Cross-Reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000.K1.12	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-28-02, 19

Learning Objective: See Attached (As available)

Question Source:
- Bank # 1347
- Modified Bank # ______ (Note changes or attach parent)
- New ______

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge L
- Comprehension or Analysis ______

10 CFR 55 Content:
- 55.41 7
- 55.43 ______

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	1347	01	04/30/2007		None	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	

Topic Area	Description
Systems	COR002280200 1050A, COR002280200 1050B Standby Gas System

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives
COR002280200 1050A Describe the interrelationships between SGT and the following: Reactor Building Ventilation System
COR002280200 1050B Describe the interrelationships between SGT and the following: Primary Containment

Related References
COR0022802

Related Skills (K/A)	
261000.K1.12	Knowledge of the physical connections and/or cause/effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Primary containment purge system: Plant-Specific (3.1 / 3.2)

QUESTION: 2

Where does the Standby Gas Treatment System line up to take a suction on an automatic initiation due to a refueling accident when the plant is in MODE 5?

a. Reactor building exhaust plenum and the primary containment exhaust ventilation line.

b. HPCI gland steam condenser exhauster and the HPCI room area.

c. Reactor building exhaust plenum and the HPCI gland steam condenser exhauster.

d. Primary containment exhaust ventilation line and the HPCI room area.

ANSWER: 2

a. Reactor building exhaust plenum and the primary containment exhaust ventilation line.

Explanation:

From Student Text
Automatic Initiation - The SGT system can be automatically started on either a high drywell pressure ($\leq$ 1.84 psig) or low-low reactor water level ($\geq$ -42 inches) initiation signal or high radiation in the exhaust plenum initiation ($\leq$ 49 mR/hr).

This signal is caused by a Group 6 containment isolation signal. Both SGTS fans will start and their respective inlet, outlet, and dilution air supply valves will open. The Group 6 isolation isolates the Reactor Building by closing the MG set ventilation valves, tripping the Reactor Building supply and exhaust fans and by isolating the normal ventilation. The SGTS suction from the Reactor Building Exhaust Plenum and SGTS room air valves draw air from the Reactor Building through the two parallel filter trains, to the fans, and then through the differential pressure control valves to the Elevated Release Point.

Distractors:

b. HPCI gland steam condenser exhauster and the HPCI room area. This is incorrect because it is only aligned here on a HPCI run with the main suction coming from the Building.

c. Reactor building exhaust plenum and the HPCI gland steam condenser exhauster. This is incorrect because it is only manually aligned here on a HPCI run with the main suction coming from the Building.

d. Primary containment exhaust ventilation line and the HPCI room area. This is incorrect because it is only manually aligned here on a HPCI run with the main suction coming from the Building.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001.K2.01	
	Importance Rating	3.3	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR001-01-01

32

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 4
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
3	00	09/21/10		AC Electrical Distribution	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	N

Topic Area	Description
Systems	List the power supplies for Off-Site sources.

Related Lessons
COR0010101 OPS AC Electrical Distribution

Related Objectives
LO 7.a. State the electrical power supplies to the following: Off-Site Sources of Power

Related References
COR0010101R32-S-OPS AC ELECTRICAL DISTRIBUTION P&ID Burns and Rowe 3001 10CFR55.41 b (4)

Related Skills (K/A)	ROI	SROI
262001.K2.01 Knowledge of electrical power supplies to the following: (CFR: 41.7) Off-site sources of power (3.3 / 3.6)	3.3	3.6

QUESTION: 3

What are the electrical power supplies to the Cooper Station Off-Site Power Sources?

a. 161KV line from Auburn; and 69KV line from OPPD only.

b. 12.5KV South Underground Line; and the 69KV line from OPPD only.

c. Auto Transformer via 345KV switchyard or the 161KV line from Auburn; and 69KV line from OPPD

d. Auto Transformer via 345KV switchyard; and 69KV line through the Corn Field Substation to the 12.5KV North Overhead Line.

ANSWER: 3

c. Auto Transformer via 345KV switchyard or the 161KV line from Auburn; and 69KV line from OPPD

Explanation:

The Start-up Transformer is energized from the Auto Transformer via OCB-1604 or from the 161KV line via OCB-1606, The Emergency Transformer is energized by the 69KV line from Omaha.

Distractors:

a. 161KV line from Auburn; and 69KV line from OPPD only. The normal feed into the station is from the 345KV switchyard was omitted from this option.

b. 12.5KV South Underground Line; and the 69KV line from OPPD only. Some of the 12.5 KV system is powered from the South Underground Line, but it is not considered part of the power supplies to the station.

d. Auto Transformer via 345KV switchyard; and 69KV line through the Corn Field Substation to the 12.5KV North Overhead Line. The 161 KV line was omitted and the North Overhead does not count as a power supply to the station.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215004.K2.01	
Importance Rating	2.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-30-02, 12

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge L
- Comprehension or Analysis

10 CFR 55 Content
- 55.41 7
- 55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
4	00	09/21/2010		AC Electrical Distribution	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	N

Topic Area	Description
Systems	What are the power supplies to the SRM Channels and Detectors?

Related Lessons
COR0023002R12-S-Source Range Monitor

Related Objectives
8.b Predict the consequences a malfunction of the following would have on the SRM system: 24/48 VDC power

Related References
COR0023002R12-S-Source Range Monitor 10CFR55.41 b.(7)

Related Skills (K/A)	ROI	SROI
215004.K2.01Knowledge of electrical power supplies to the SRM channels/detectors	2.6	2.8

QUESTION: 4

What are the power supplies to the SRM Channels/detectors?

SRM Channels…

a. A & B are powered from Division I $\pm$24VDC System and C & D are powered from Division II $\pm$24VDC System.

b. A & C are powered from Division I $\pm$24VDC System and B & D are powered from Division II $\pm$24VDC System.

c. A & B are powered from Division I 125 VDC System and C & D are powered from Division II 125 VDC System.

d. A & C are powered from Division I 125 VDC System and B & D are powered from Division II 125 VDC System.

ANSWER:

b. A & C are powered from Division I $\pm$24VDC System and B & D are powered from Division II $\pm$24VDC System.

Explanation:

From the Student Text for Lesson COR002-30-02 the Power Supplies for the SRM subsystem is powered from the $\pm$ 24 VDC system. Channel A and C are powered from Div 1 and channel B and D are powered from Div 2.

Distractors:

a. A & B are powered from Division I $\pm$24VDC System and C & D are powered from Division II $\pm$24VDC System is incorrect because SRM “B” and “D” are DIV II SRMs and “A” and “C” are the DIV I SRMs.

c. A & B are powered from Division I 125 VDC System and C & D are powered from Division II 125 VDC System. This is the normal power supply form 125 VDC DIV I components and logics, however the SRMs are powered from the 24 VDC system.

d. A & C are powered from Division I 125 VDC System and B & D are powered from Division II 125 VDC System. This is the normal power supply form 125 VDC DIV II components and logics, however the SRMs are powered from the 24 VDC system.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215005.K3.07	
Importance Rating	3.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-01-02
22

Learning Objective: See Attached (As available)

Question Source:
Bank # 6137
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 6
55.43 ______

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
05 6137	01	12/19/2005		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0022402, 100% power with 26-27 selected. APRM E is 80% while all other APRMs are 100%.

Related Lessons
COR0020102 OPS AVERAGE POWER RANGE MONITOR COR0022402 OPS ROD BLOCK MONITOR SKL0124224 ROD BLOCK MONITOR SYSTEM

Related Objectives	
COR0020102001070E	Given a specific APRM malfunction, determine the effect on any of the following: Rod Block Monitoring System (RBM)
COR0022402001040A	Describe the RBM design features and/or interlocks that provide for the following: Prevent control rod withdrawal.
SKL012422400A030A	Given plant conditions, predict changes in the following Rod Block Monitor System components/parameters: Trip reference

Related References

Related Skills (K/A)		ROI	SROI
215005.K3.07	Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: (CFR: 41.7 / 45.4) Rod block monitor: Plant-Specific (3.2 / 3.3)		

QUESTION: 5 6137

The plant is operating at 100% power with control rod 26-27 selected. Average Power Range Monitor (APRM) channel E is reading 80% while all other APRMs are reading 100%.

What effect will this have on Rod Block Monitor (RBM) channel A?

RBM channel A will…

a. initiate a Flow Reference Off-Normal rod block.

b. immediately initiate an RBM Downscale rod block.

c. enforce a less-than-conservative RBM Upscale rod block.

d. automatically transfer to APRM C as its reference APRM.

ANSWER: 5 6137

c. enforce a less-than-conservative RBM Upscale rod block.

Explanation:

The APRM's provide a reference power level for use in the Rod Block Monitor system. If the APRM used to set up the RBM trip references is indicating ≥ 30% (*) power, the RBM is zeroed, and RBM outputs are bypassed. If the reference APRM is bypassed, the reference signal is automatically provided by a second APRM. The reference APRM for RBM Channel A is APRM Channel E, with alternate APRM Channel C. The reference APRM for RBM Channel B is APRM Channel B, with alternate APRM Channel D.

(*) The actual set point number is≥27.5%, but the Tech Spec Basis number for assuming RBM will mitigate the consequences of a RWE event with a peripheral control rod not selected is ≥30%.

When the reference APRM reads low the RBM assumes that power is lower than it actually is and will use the trip references for that power level band to assign rod blocks. The reference trip for 80% power is less than the one for 90 and above.

Distractors:

a. The Rod Block Monitor uses Recirc Flow for the Flow Reference Off-Normal rod block, not the Reference APRM.

b. The RBM looks at the reference APRM signal to determine if it is downscale from it or not. So in this case it would not be.

d. APRM C is not the alternate for RBM A.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000.K3.02	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-18-02
17

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6		00	12/15/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	How do Reactor pressure and RCIC speed response to flow controller failure?

Related Lessons
COR0021802 RCIC

Related Objectives	
COR0020602001050A	Describe the Core Spray system design features and/or interlocks that provide for the following: Prevention of over pressurization of Core Spray piping

Related References
COR0021802

Related Skills (K/A)	
217000.K3.02	Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on Reactor vessel pressure

QUESTION: 6

The Reactor Core Isolation Cooling (RCIC) System is in the pressure control mode of operation in accordance with Procedure 2.2.67.1 Pressure Control with RCIC Hard card, following a reactor scram thirty minutes ago, with the following alignment:

- RCIC-FIC-91, RCIC flow controller is in Automatic.
- RCIC-FIC-91, RCIC flow controller Set tape is set to 400 GPM.

Shortly after the alignment was established the following occurs:

- RCIC-FT-58, discharge flow transmitter has failed high such that the flow sensed by the RCIC-FIC-91 is 500 gpm irrespective of actual RCIC flow.

How do Reactor pressure and RCIC speed response to this failure?

	Reactor Pressure	RCIC Speed
a.	slowly rises	rises and stabilizes at 5000 RPM
b.	slowly rises	lowers to idle speed
c.	slowly lowers	rises and stabilizes at 5000 RPM
d.	slowly lowers	lowers to idle speed

ANSWER: 6

b. slowly rises lowers to idle speed

Explanation:

The maximum demand (500 gpm) to the RCIC flow controller will cause FIC-91 to run speed at minimum, idle speed.

Distractors:

a. It is true that reactor pressure will rise because the RCIC Turbine is no longer providing pressure control for the vessel. The speed will lower not rise.

c. Reactor pressure will rise because the RCIC Turbine is no longer providing pressure control for the vessel, and the speed will lower not rise.

d. Reactor pressure will rise because the RCIC Turbine is no longer providing pressure control for the vessel.

Examination Outline Cross-Reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001.K4.01	
	Importance Rating	3.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-06-02, 21

Learning Objective: See Attached (As available)

Question Source:
Bank # 1691
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43 ____

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	1691	00	08/04/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001050A	Describe the Core Spray system design features and/or interlocks that provide for the following: Prevention of over pressurization of Core Spray piping

Related References
COR0020602

Related Skills (K/A)	
209001.K4.01	Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) Prevention of over pressurization of core spray piping (3.2 / 3.4)

QUESTION: 7

Which interlock prevents the Core Spray piping from being overpressurized?

a. When the suction valve (MO-7 A/B) is closed, the pump motor supply breaker must be racked out.

b. When reactor pressure > 436 psig, both injection valves may be opened at the same time if an initiation signal is present.

c. When reactor pressure < 436 psig, both injection valves may be opened at the same time if inboard injection valve (MO-12 A/B) is opened first.

d. When reactor pressure > 436 psig, the outboard injection valve (MO-11 A/B) can be opened if inboard injection valve (MO-12 A/B) is closed.

ANSWER: 7

d. When reactor pressure > 436 psig, the outboard injection valve (MO-11 A/B) can be opened if inboard injection valve (MO-12 A/B) is closed.

Explanation:

The Injection piping up stream of the MO-11 A/B is low pressure piping and cannot withstand the pressure of the vessel if both the 11 and the 12 were opened at the same time while the Reactor was operating at rated pressure.

Distractors:

a. It is true that the Core Spray Pumps do not trip on a loss of suction valve alignment. However this is a low pressure concern not an overpressure concern.

b. The pressure is correct however one valve can be opened at a time.

c. The pressure is incorrect the interlock is for pressures >436 psig.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002.K4.01	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR001-01-01
32

Learning Objective: See Attached (As available)

Question Source:
Bank # 1538
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
8 1538	02	09/27/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	5	Multiple Choice	N

Topic Area	Description
Systems	How does the NBPP react to a loss of normal power?

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001090G	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Transfer from preferred power to alternate power supplies

Related References	
LP	COR0010102

Related Skills (K/A)		ROI	SROI
262002.K4.01	Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Transfer from preferred power to alternate power supplies (3.1/3.4) (3.1 / 3.4)		

QUESTION: 8

How does the No-Break Power Panel respond to a loss of its normal power supply?

The NBPP will transfer from...

a. Inverter 1A to Inverter 1B.

b. Inverter 1A to MCC-R (115 V).

c. MCC-R (115V) to Inverter 1A.

d. MCC-R (155V) to selected Critical Distribution Panel.

ANSWER: 8

b. Inverter 1A to MCC-R (115V)

Explanation:

Power to the No-Break Power Panel (NBPP) #1 is normally supplied from 250 VDC bus 1A through inverter 1A and a static switch.

An emergency (alternate) AC power source for the NBPP #1 is provided from MCC-R through a step-down transformer in the event that inverter 1A fails.

Distractors:

a. The NBPP will not automatically aligned to Inverter B

c. The NBPP is normally aligned to Inverter A therefore MCC_R is the backup power supply, not the other way around.

d. The NBPP is normally aligned to Inverter A therefore MCC_R is the backup power supply, not the other way around.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000.K5.02	
	Importance Rating	3.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-23-02 27

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
09	00	09/27/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	What effect will taking the control switch for the RHR-MO-66B closed for 30 seconds have on core cooling and heat exchanger flow?

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives
COR0022302001060D Given an RHR control manipulation, predict and explain changes in the following: Reactor parameters (level, pressure, temperature)

Related References
COR002-23-02 RHR Lesson Rev 27

Related Skills (K/A)	ROI	SROI
203000.K5.02 Knowledge of the operational implications of Core cooling methods as they apply to RHR/LPCI: INJECTION MODE:	3.5	3.7

QUESTION: 9

The plant is operating at 100% power when a LOCA occurs. The following conditions exist:

- Rx level is steady in the normal band and is being controlled by B loop of RHR
- Drywell pressure is 5.5 psig and steady
- RHR SW Booster Pump 1D is started

Four (4) minutes after the LOCA signal occurs; the control room operator places the control switch for the RHR-MO-66B Heat Exchanger Bypass valve in the closed position for 30 seconds. What effect will this action have on core cooling and heat exchanger flow?

	Core Cooling	Heat Exchanger Flow
a.	remains constant	remains constant
b.	remains constant	increases
c.	increases	remains constant
d.	increases	increases

ANSWER: 9

d. increases increases

Explanation:

During LPCI operation, the RHR pumps discharge through the RHR Heat Exchanger and heat exchanger bypass valve. The heat exchanger bypass valve is initially interlocked open for 2 minutes in order to maximize the flow rate into the reactor vessel.

Distractors:

a. Core Cooling cannot remain constant because cooler water is being injected into the vessel, as more flow is directed through the RHR Heat Exchanger. A candidate might miss this if they misapplied the 5 minute time interlock for the injection valve with the Heat Exchanger Bypass valve 2 minute interlock.

b. Core Cooling cannot remain constant because cooler water is being injected into the vessel, as more flow is directed through the RHR Heat Exchanger.

c. Core Cooling increases because cooler water is being injected into the vessel, as more flow is directed through the RHR Heat Exchanger and therefore cannot remain constant.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003.K3.01	
	Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-12-02, 12

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10		02	09/27/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	2	Multiple Choice	

Topic Area	Description
Systems	How will this affect the operation of “A” IRM versus the others when subjected to the same neutron field?

Related Lessons
COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives
COR0021202001060D Given a specific IRM malfunction, determine the effect on any of the following: Reactor power indication

Related References
COR0021202 Rev 12 Intermediate Range Monitor

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: 10

The “A” IRM was installed with half the argon fill pressure then the other seven detectors.

How will this affect the operation of “A” IRM versus the others when subjected to the same neutron field?

IRM “A” is…

a. more sensitive and therefore will read higher than the others.

b. less sensitive and therefore will read higher than the others.

c. more sensitive and therefore will read lower than the others.

d. less sensitive and therefore will read lower than the others.

ANSWER: 10

d. less sensitive and therefore will read lower than the others.

Explanation:

From the Student Text COR002-12-02 Rev 12.
The IRM detector is very similar to the SRM detector, but is **less sensitive** due to the following major differences; the IRM detectors have: Less uranium, **Lower argon gas pressure**, Lower operating voltage.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	205000.K6.04	
Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Abn. Procedure 2.4SDC 12

Learning Objective: See Attached (As available)

Question Source:
Bank # 7763
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	7763	01	02/21/2005		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030K	Describe RHR System design feature(s) and/or interlocks which provide for the following: Low reactor water level isolation
COR0022302001050B	Briefly describe the following concepts as they apply to the RHR system: Valve operation
COR0022302001080K	Predict the consequences a malfunction of the following will have on the RHR system: Reactor water level

Related References	
10CFR55.41	Written examinations: Operators
2.4SDC	Shutdown Cooling Abnormal

Related Skills (K/A)	
205000. K6.04	Knowledge of the effect that a loss or malfunction of Reactor water level will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): 3.6 3.6

QUESTION: RO 11 7763

The plant is shutdown with a cooldown in progress.

- Reactor pressure is 50 psig (stable)
- RPV level is +45 inches (NR stable)
- RHR loop B suction path is lined up for Shutdown Cooling

RPV level then lowers to -150 inches (WR) due to a recirculation suction line break.
What is the <u>first</u> control room valve manipulation required to establish injection with LPCI Loop B?

a. OPEN RHR-MO-27B, LPCI Injection Valve.

b. CLOSE RHR-MO-15B and 15D, S/D Cooling Suction Valves.

c. CLOSE RHR-MO17 and 18, S/D Cooling Suction Isolation Valves.

d. OPEN RHR-MO-13B and 13D, Torus Cooling Suction Valves.

ANSWER: RO 11 7763

b. CLOSE RHR-MO-15B and 15D, Shutdown Cooling Suction Valves.

Explanation:

With the SDC suction valves open the loss of reactor level results in a SDC isolation. In order to establish B loop LPCI injection the suction has to be realigned to the torus. MO-15B and 15D must be closed before 13B and 13D can be reopened.

Distractors:

a. is incorrect as this valve will open and will not close on the SDC isolation.

c. is incorrect because this action does not need to be performed by the operator.

d. is incorrect because the torus suction valves cannot be opened unless the SDC suctions are closed. The PCIS isolation will close these valves.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	400000.K6.07	
Importance Rating	2.7	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-19-02
20

Learning Objective: See Attached (As available)

Question Source:
Bank # 18245
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
12 18245	01	03/20/2005		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0021902, REC pump response following Loss of Off-site power

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001070D	Predict the consequences a malfunction of the following would have on the REC system: Loss of Normal AC power

Related References	
CFR	10CFR55.41

Related Skills (K/A)		ROI	SROI
40000.K6.07	Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: (CFR: 41.7 / 45.7) Breakers, relays, and disconnects (2.7 / 2.8)		

QUESTION: 12

The plant is operating normally at full power with the following REC alignment:

- REC pumps 1B, 1C, and 1D are running
- REC pumps 1B and 1C are in STANDBY

The plant experiences a loss of all offsite power.

What is the expected REC pump status 1 minute after the loss of all offsite power?

a. Only pump 1C will be running.

b. Only pump 1D will be running.

c. Only pumps 1B and 1C will be running.

d. Pumps 1B, 1C, and 1D will ALL be running.

ANSWER: 12

c. Only pumps 1B and 1C will be running.

Explanation:

Following the loss of all offsite power, under-voltage on 4160V bus 1F and 1G will start DG-1 and DG-2 and will result in a loss of power to MCC-K and to MCC-S and the subsequent trip of all running REC pumps. When DG-1 and DG-2 have reached rated speed and voltage and breakers EG-1 and EG-2 have closed, then REC pumps 1C and 1B will auto restart 20 seconds after MCC-K and MCC-S are re-energized via bus 1F and 1G.

Distractors:

a. is incorrect. Because pump 1B will auto restart.
b. is incorrect. Because pump 1C will auto restart.
d. is incorrect. Because pump 1D will trip and will not automatically restart.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000.A1.04	
	Importance Rating	2.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-08-02
(Attach if not previously provided)
(including version/revision number) 24

Learning Objective: See Attached (As available)

Question Source: Bank # 19288
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 8
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	19288	02	12/17/2004		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020802, Loss of SW to DG with Auto start present

Related Lessons
COR0020802 OPS DIESEL GENERATORS

Related Objectives	
COR0020802001060H	Describe the interrelationship between Diesel Generators and the following: Service Water System
COR0020802001140C	Given plant conditions, determine if the following should occur: Diesel Generator trip
COR0020802001010G	State the purpose of the following items related to the Diesel Generators: Cooling Water subsystem

Related References
NONE

Related Skills (K/A)
264000.A1.04 Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Crank case temperature and pressure (2.6/2.7)

QUESTION: 13 19288

The following conditions exist:

- A LOCA has occurred.
- DG1 has been supplying its loads for 5 minutes.
- SW-2797A (Service water admission valve to DG1) remains closed.

How is DG1 affected?

a. DG1 trips on low lube oil pressure.

b. DG1 trips on high jacket water temperature.

c. DG1 continues to operate using Division II service water supply.

d. DG1 continues to operate with high lube oil and jacket water temperatures.

ANSWER: 13 19288

d. DG1 continues to operate with high lube oil and jacket water temperatures.

Explanation:

A loss of service water will cause jacket water temperatures to rise and the diesel will continue to run since this trip is bypassed by the LOCA signal.

Distractors:

a. DG oil temperatures will rise, but only a high oil temperature alarm is provided no trip. The low lube oil pressure trip is also bypassed.

b. The DG jacket water high temperature trip is bypassed with the diesel started by a LOCA signal.

c. The DG service water header admission valve is on the inlet to the cooling systems and downstream of the other division supply.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000.A1.08	
	Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s)	Procedure 2.2.33.1;	Lesson COR002-11-02
(Attach if not previously provided)		
(including version/revision number)	28	26

Learning Objective: See Attached (As available)

Question Source:	Bank #	3744	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content	55.41	5
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	3744	04	09/28/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	SKL0124211, COR0021102 High Pressure Coolant Injection

Related Lessons
COR0021102 OPS High Pressure Coolant Injection System

Related Objectives	
COR0021102001120C	Given plant conditions, determine if the following HPCI actions should occur: Minimum flow valve change of position
COR0021102001050E	Describe the interrelationship between HPCI and the following: ECST
COR0021102001050F	Describe the interrelationship between HPCI and the following: Suppression chamber

Related References	
2.2.33.1	High Pressure Coolant Injection System Operations

Related Skills (K/A)	
206000.A1.08	Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: System lineup: (4.1* 4.0)

QUESTION: 14 3744 (1 point(s))

The High Pressure Coolant Injection (HPCI) System is started in RPV PRESSURE CONTROL MODE at a flow rate of 2000 gpm and a discharge pressure of 900 psig.

The Reactor Operator then lowers the FLOW CONTROLLER HPCI-FIC-108 to establish a flow rate of 400 gpm for 12 minutes.

What is the effect on Suppression Pool level **AND** the Emergency Condensate Storage Tanks (ECST) level due to lowering the HPCI flow to this new value?

a. Suppression Pool level will rise.
ECST level will rise.

b. Suppression Pool level will rise.
ECST level will lower.

c. Suppression Pool level will lower.
ECST level will remain the same.

d. Suppression Pool level will remain the same.
ECST level will remain the same.

ANSWER: 14 3744

b. Suppression Pool level will rise.
ECST level will lower.

Explanation:

In the RPV PRESSURE CONTROL MODE alignment, HPCI suction is aligned to the ECST with a return flow path to the ECST. However the operator's action to lower the flow controller to 400 gpm causes the minimum flow valve to open and it returns to the torus. Water is then diverted from the ECST to the TORUS.

Distractors:
a, c, & d. are incorrect because the Full Flow Alignment has HPCI suction aligned to the ECST and at 400 gpm the min flow valve to the Torus is OPEN, diverting ECST water to the TORUS.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000.A2.08	
	Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.1.5
64

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 2
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
15	00	09/29/10		License Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	Scram and Startup FCVs response on low level

Related Lessons
COR0022102 OPS Reactor Protection System

Related Objectives
COR0022102001090C Predict the consequences a malfunction of the following would have on the RPS system: Nuclear boiler instrumentation

Related References
COR0022102 OPS Reactor Protection System Rev20

Related Skills (K/A)	ROI	SROI
212000.A2.08 Ability to (a) predict the impacts of Low reactor level on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (4.1*/4.2*)	4.1	4.2

QUESTION: 15

The reactor is at 100% power with the Startup FCVs in Manual due to a Triconex failure when the following occurs:

- RPV water level starts lowering.
- RFC-LI-94A, B & C read 0 inches.

What is the status of the RPS Trip System **and** what must be performed on the RVLC HMIs for the Startup valves FCV-11AA or FCV-11BB to allow feeding the vessel to recover reactor water level?

	RPS TRIP Logic	FCV-11AA or FCV-11BB Action
a.	Tripped	Press AUTO button on the Startup Master Level
b.	Tripped	Press GREEN UP ARROW on the Startup Master Level
c.	Not Tripped	Press AUTO button on the Startup Master Level
d.	Not Tripped	Press GREEN UP ARROW on the Startup Master Level

ANSWER: 15

b. Tripped Press GREEN UP ARROW on the Startup Master Level

Explanation:

With all three level indicators RFC-LI-94A, B & C reading 0 inches would indicate that the NBI-LIS-101A through D would read the same, as they come off the same reference and variable instrument legs. The NBI-LIS-101A, B, C, D feed the RPS system to produce a Reactor Scram on low level at 3 inches Tech Spec and 10 inches actual setpoint. For the Startup FCVs, they are controlled by the FWLCS and if in AUTO on a reactor scram, will regulate to control RPV level when Level Set-Down is enabled. In this setup, they will not, since they are in manual. IAW Procedure 2.1.5 Attachment 3 REACTOR WATER LEVEL CONTROL; step 1.3.6 Adjust STARTUP MASTER controller using UP/DOWN arrows or RAMP FUNCTION to adjust LEVEL SETPOINT as desired.

Distractors:

a. RPS Trip Logic will be tripped, the failure prevents selecting AUTO to control the STARTUP FCVs to prevent overfeeding.

c. RPS Trip Logic will be tripped.

d. RPS Trip Logic will be tripped, the failure prevents selecting AUTO to control the STARTUP FCVs to prevent overfeeding.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000.A2.04	
	Importance Rating	2.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-28-02; Ann Proc. 2.3_K-1
19 12

Learning Objective: See Attached (As available)

Question Source:
Bank #
Modified Bank # 1164 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
16	00	09/29/10		License Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	What is the major impact to the system **and** what must be done to mitigate the consequences?

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001100I	Predict the consequences of the following on the Standby Gas Treatment system: High train moisture content

Related References
COR0022802 OPS Standby Gas Treatment System R19 AP 2.3K-1 Alarm Procedure 2.3_K-1 Rev 12

Related Skills (K/A)	ROI	SROI
261000.A2.04 Ability to (a) predict the impacts of High train moisture content on the STANDBY GAS TREATMENT 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) (2.5 / 2.7)	2.5	2.7

QUESTION: 16

The plant is operating at 100% power with the “A” SGT train in service to support HPCI full flow surveillance when the following occurs:

- Annunciator K-1/A-2 SGT A HIGH MOISTURE alarms

What is the major impact to the system due to this condition **and** what must be done to mitigate the consequences of the high moisture condition?

	Impact	Action
a.	Reduced iodine adsorption in the charcoal filter.	Check SGT-DPIC-546 for proper operation.
b.	Reduced iodine adsorption in the charcoal filter.	Start SGT B train and secure SGT A.
c.	Potential fire hazard in the Inlet High Efficiency filter	Check SGT-DPIC-546 for proper operation.
d.	Potential fire hazard in the Inlet High Efficiency filter	Start SGT B train and secure SGT A.

ANSWER: 16

b. Reduced iodine adsorption in the charcoal filter. Start SGT B train and secure SGT A.

Explanation:

From the student study material: Excessive moisture or organic materials (such as lubricants) will reduce the iodine adsorption capability of the charcoal filter if these are not previously removed by the demister, heater, or filter 1.

When Alarm K-1 / A-2 SGT A HIGH MOISTURE alarms the annunciator procedure directs them to perform or check the following:

1. OPERATOR OBSERVATION AND ACTION
   1 .1 Check heater operation.
   1 .2 Start SGT B train and secure SGT A per Procedure 2 .2.73.

Distractors:

a. Reduced iodine adsorption in the charcoal filter is the correct item of concern but checking SGT-DPIC-546 for proper operation is the wrong action to take.

c. Potential fire hazard in the Inlet High Efficiency filter is the wrong component. There is a higher potential for fire in the charcoal filter rather than the high efficiency filter. Also checking SGT-DPIC-546 for proper operation is also wrong action.

d. Potential fire hazard in the Inlet High Efficiency filter is the wrong component. There is a higher potential for fire in the charcoal filter rather than the high efficiency filter. But the action is correct.

Modified question: Bank question that was modified - 1164

With regard to the Standby Gas Treatment (SGT) system, which one of the following is a consequence of an inefficient Moisture Separator?

a. Overload trip of the associated SGT fan.

b. Excessively high water level in the Z sump.

c. Reduced iodine adsorption in the charcoal Filter.

d. Potential fire hazard in the Inlet High Efficiency Filter.

ANSWER: 1 1164

c. Reduced iodine adsorption in the Charcoal Filter.

REFERENCE: STCOR002-28-02, page 14, section II.F.2, rev. 10.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005.A3.02	
	Importance Rating	3.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 2.3_9-5-1 Panel 9-5 - Annunciator 9-5-1
(Attach if not previously provided)
(including version/revision number) 24

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 2
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17		00	01/11/2011		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020102, AVERAGE POWER RANGE MONITOR

Related Lessons
COR0020102 OPS AVERAGE POWER RANGE MONITOR

Related Objectives	
COR0020102001050D	Describe the interrelationships between the Average Power Range Monitor System and the following: Local Power Range Monitoring System (LPRM)

Related References	
2.3_9-5-1	Panel 9-5 - Annunciator 9-5-1

Related Skills (K/A)
215005.A3.02 Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Full core display (3.5 / 3.5)

QUESTION: 17

The plant is performing a startup and power ascension from 1% to 100% power. As power is raised, the LPRM Low Alarm lights begin clearing on the Full Core Display.

What approximate average APRM power level will the last LPRM low light and LPRM Low Alarm Clear on Panel 9-5?

a. 3% power.

b. 9% power.

c. 15% power.

d. 40% power.

ANSWER: 17

d. 40% power.

Explanation:

The LPRMs are widely distributed throughout the core and while some will start clearing their low alarms at low APRM Average power levels, it take a power increase to approximately 40% before the lowest reading LPRM will finally clear on the full core display and the Annunciator for the LPRM low to clear.

Distractors:

a. This is the power level of the individual LPRM to start clearing, not the last one to clear

b. This is the power were approximately one quarter of the LPRMs will be cleared.

c. This is the power level were approximately half of the LPRMs will be cleared.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000.A3.04	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-18-02
17

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis H

10 CFR 55 Content
- 55.41 7
- 55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
18	00	10/04/10		License Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	Without operator action, how is RCIC flow affected, if at all?

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling

Related Objectives	
COR00218020011300	Briefly describe the RCIC system response to an initiation signal when in a normal standby alignment.

Related References		
COR0021802	OPS Reactor Core Isolation Cooling	Rev 17

Related Skills (K/A)	ROI	SROI
217000.A3.04 Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: (CFR: 41.7 / 45.7) System flow (3.6 / 3.5)	3.6	3.5

QUESTION: 18

The reactor is at 100% power with RCIC in a full flow test mode using the test potentiometer at 300 gpm, when a loss of feedwater occurs.

- Reactor scrams and water level lowers to - 45 inches Wide Range.

Without operator action, how is RCIC flow affected, if at all?

RCIC flow…

a. remains at 300 gpm in the test mode.

b. remains at 300 gpm however that flow is directed to the vessel.

c. increases to 400 gpm however RCIC remains in the test mode.

d. increases to 400 gpm however that flow is directed to the vessel.

ANSWER: 18

d. increases to 400 gpm however that flow is directed to the vessel.

Explanation:

Even though RCIC is in the full flow test alignment on the test potentiometer an initiation signal will cause RCIC to shift out of the test mode to a full automatic mode using the normal setpoint for flow (i.e. 400 gpm injecting into the vessel). RCIC starts on a level 2 setpoint of – 42 inches.

A Test Potentiometer on Panel 9-4 provides an additional method of controlling turbine speed. The Test Potentiometer sends a signal to open or close the governor valve as desired. To use the test potentiometer, the Test switch is placed to TEST and the Test Power switch is placed in ON. The initiation signal must be reset if present. If an initiation occurs while using the test potentiometer, flow control will be returned to the GEMAC flow controller.

Distractors:

a. The system shifts to the flow controller on an initiation signal and injects into the vessel.

b. The system shifts to the flow controller on an initiation signal and injects into the vessel.

c. The system shifts to the flow controller on an initiation signal and injects into the vessel.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	223002.A4.02	
Importance Rating	3.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) T.S. 3.3.6.1 Procedure 2.1.22
(Attach if not previously provided)
(including version/revision number) Amendment 178 55

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19		00	12/02/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	Given a set of plant conditions determine if a Group 3 should occur

Related Lessons
COR0020302 OPS CONTAINMENT

Related Objectives
21b. Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations

Related References
2.1.22 Recovering From A Group Isolation

Related Skills (K/A)	
223002.A4.02	Ability to manually operate and/or monitor in the control room: Manually initiate the system (3.9/3.8)

QUESTION: 19

The Plant is operating at 100% power when the following events occur:

- RWCU System Heat Exchanger Room temperature is 185°F and going up at 1°F per minute.

When will the RWCU system isolate and what actions should be taken if it does not?

a. in 7 minutes; Close RWCU-MO-15 **Only**, and RWCU-MO-74 should be cracked open.

b. in 7 minutes; Close both RWCU-MO-15 and RWCU-MO18, and RWCU-MO-74 should be cracked open.

c. in 10 minutes; Close RWCU-MO-15 **Only**, and RWCU-MO-74 should be cracked open.

d. in 10 minutes; Close both RWCU-MO-15 and RWCU-MO18, and RWCU-MO-74 should be cracked open.

ANSWER: 19

d. in 10 minutes; Both RWCU-MO-15 and RWCU-MO18, and RWCU-MO-74 should be cracked open.

Explanation: with a temperature rise in the RWCU System Heat Exchanger Room the PCIS system should react and cause an isolation of the RWCU system when temperature reaches 195°F in accordance with Procedure 2.1.22. This is the Tech Spec Allowable Value. The rate of rise is given as 1°F/minute so it will take ten more minutes to take the room temperature from 185°F to 195°F.

Distractors:

a. This time is too short, an isolation will not occur until the temperature in the room reaches 195°F. If that does not occur then the operator is required to take manual action to cause the failed automatic action to happen at the setpoint. Both isolation valves should be closed because the auto action would have closed both valves.

b. This time is too short, an isolation will not occur until the temperature in the room reaches 195°F. If that does not occur then the operator is required to take manual action to cause the failed automatic action to happen at the setpoint. Both isolation valves should be closed because the auto action would have closed both valves.

c. This time is correct however the operator is required to take manual action to cause the failed automatic action to happen at the setpoint. Both isolation valves should be closed because the auto action would have closed both valves.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000.K1.03	
	Importance Rating		2.6

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-07-02, Abnormal Proc. 2.4RXLVL

(Attach if not previously provided)
(including version/revision number) 29 24

Learning Objective: See Attached (As available)

Question Source: Bank # 15139
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	16412	01	06/26/2008	05/23/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
Systems	Monitor RVLCS when 1 Steam Flow instrument is removed from service.

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR0023202001010C	State the purpose of the following items related to the Reactor Vessel Level Control System: Steam flow instruments
COR0023202001020A	Describe the interrelationship between RVLC and the following: Main Steam
COR0023202001050B	Briefly describe the following concepts as they apply to the RVLC system: Steam flow/Feed Flow Mismatch

Related References	
2.4RXLVL	RPV Water Level Control Trouble

Related Skills (K/A)	
259002.A4.07	Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from automatic to manual mode (3.8)

QUESTION: 20 16412 (1 point(s))

The plant is at 75% power. The Reactor Level Control system is maintaining RPV level at +35 inches in three (3) element control. HMI shows mFT0051C_INVALID, MS Flow Ch. C INVALID.

If the operator were to then bypass MS-PT-56 Turbine 1st stage pressure, what would be the status of the RVLCS?

a. RVLCS would remain in 3 element control with level at +35 inches.

b. RVLCS would transfer to single element control with level at +35 inches.

c. RVLCS would remain in 3 element control and RPV level LOWERS and stabilizes at approximately +26 inches.

d. RVLCS would transfer to single element control and RPV level LOWERS and stabilizes at approximately +26 inches.

ANSWER: 20 16412

b. RVLCS would transfer to single element control with level at +35 inches.

Explanation:

Level will transfer to single element due to 1 invalid steam flow instrument when the turbine first stage pressure instrument is bypassed. Level will remain at 35”.

Distractors:

a, c, d: RVLCS will transfer to single element. Level with the previous RVLCS lowered until it stabilizes at 26 inches.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.4.45	
	Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 2.4SRV 2.3_9-3-1 Annunciator Panel Procedure

(Attach if not previously provided)
(including version/revision number) 11 28

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21		00	01/11/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0021602, What temp corresponds to the pressure the amber light is lit?

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001040A	Given a Nuclear Pressure Relief system component manipulation, predict and explain the changes in the following parameters: Tail pipe temperatures
COR0021602001060D	Briefly describe the following concepts as they apply to NPR: Tail pipe temperature monitoring

Related References
2.4SRV 2.3_9-3-1 Annunciator Panel Procedure

Related Skills (K/A)
239002 SRV 2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12) IMPORTANCE RO 4.1 SRO 4.3.

QUESTION: 21

The plant is operating at 100% power when the MSIV automatically close. The reactor scrams and several SRVs lift to relieve pressure.

Several minutes into the transient, HPCI is placed in pressure control and is controlling RPV pressure at 1000 psig steady. The RO reviews of the annunciators that are still in and which have cleared and notes that Panel 9-3-1/C-1 Safety/Relief Valve Leaking, is still in and that 9-3-1/A-2 Relief Valve Open is cleared.

What action is required?

a. check SRV tailpipe temperatures only.

b. enter 2.4SRV and check SRV tailpipe temperatures.

c. enter 2.4SRV and cycle the SRVs with high tailpipe temperatures.

d. enter 2.4SRV, cycle the SRVs with high tailpipe temperature, then pull the fuses for those SRVs.

ANSWER: 21

b. enter 2.4SRV and check SRV tailpipe temperatures.

Explanation:

The tailpipe temperatures for recently opened SRVs will remain high enough to cause the leaking alarm. These temperatures take time to reduce and so long as they continue to lower and remain within the pack, this is a normal response. With that said, there are directions to enter 2.4SRV contained within the alarm card for 9-3-1/C-1.

Distractors:

a. is a correct action, because the tailpipe temperatures must be checked to ensure that they are lowering and that one or more are not leaking by. However there is an entry condition for 2.4SRV.

c. is a correct action, because the tailpipe temperatures must be checked to ensure that they are lowering and that one or more are not leaking by. However there is no reason to cycle the SRV given in the stem.

d. is a correct action, because the tailpipe temperatures must be checked to ensure that they are lowering and that one or more are not leaking by. However there is no reason to cycle the SRV or pull the fuses for it given in the stem.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.1.19	
	Importance Rating	3.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-17-02 and Proc. 2.2.1
(Attach if not previously provided)
(including version/revision number) 16 37

Learning Objective: See Attached (As available)

Question Source: Bank #
Modified Bank # 3297 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22		00	12/07/2010		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	Determine if the ADS valves should be open and how they would be displayed on SPDS.

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001050B	Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: ADS logic control
COR0021602001060A	Briefly describe the following concepts as they apply to NPR: ADS logic operation

Related References	
2.2.1	Nuclear Pressure Relief System

Related Skills (K/A)
218000 ADS 2.1.19 Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12) IMPORTANCE RO 3.9 SRO 3.8

QUESTION: 22 (1 point(s))

The following conditions have been present for 2 minutes:

- RPV water level indicates -148 inches on the wide range RPV level instruments.
- Reactor pressure is 300 psig.
- Drywell pressure is 22 psig.

Which one of the following describes the current status of the ADS valves, and how are the SRVs displayed on SPDS 10 Suppression Pool Mimic?

The ADS valves are…

a. **open.**
The ADS Valves are highlighted red and the LLS valves are highlighted red.

b. **open.**
The ADS Valves are highlighted red and the LLS valves are highlighted green.

c. **closed.**
The ADS Valves are highlighted green and the LLS valves are highlighted red.

d. **closed.**
The ADS Valves are highlighted green and the LLS valves are highlighted green.

ANSWER: 22

b. **open.**
The ADS Valves are highlighted red and the LLS valves are highlighted green.

Explanation: With Reactor Water level below the -113 inch setpoint the ADS Timers will start and 109 seconds later the ADS valves will open if there are ECCS Pumps running as indicated by a discharge pressure. With water level below the automatic initiation setpoints for the ECCS Pumps, the pumps will be running on minimum flow.

SPDS 10 Suppression Pool Mimic displays the status of the SRVs, both the ADS and the LLS valves in either green (closed) or red (open), along with the tailpipe temperatures, the suppression pool temperature and the status of both HPCI and RCIC.

Distractors:

a. ADS valve logic is satisfied and the valves are open. SPDS 10 would indicate the ADS Valves as red, but the LLS (low low set) valves would indicate green.

c. ADS valve logic is satisfied and the valves are open. SPDS 10 would indicate the ADS Valves as red, but the LLS (low low set) valves would indicate green.

d. ADS valve logic is satisfied and the valves are open. SPDS 10 would indicate the ADS Valves as red, but the LLS (low low set) valves would indicate green.

REFERENCE: COR0021602, Procedure 2.2.1; Procedure 2.4CSCS

Modified question # 3297

QUESTION: 3297

The following conditions have been present for 2 minutes:

- RPV water level indicates -148 inches on the wide range RPV level instrument.
- Reactor pressure is 300 psig.
- Drywell pressure is 22 psig.

Assume ALL equipment operates as designed.

Which one of the following describes the current status of the ADS valves, and the actions necessary to close **OR** maintain them closed?

The ADS valves are…

a. **open.**
The ADS A INHIBIT and the ADS B INHIBIT switches must be placed in INHIBIT.

b. **closed.**
The ADS A INHIBIT and the ADS B INHIBIT switches must be placed in INHIBIT.

c. **closed.**
The ADS LOGIC A TIMER and the ADS LOGIC B TIMER pushbuttons must be depressed at least every 90 seconds.

d. **open.**
The ADS A INHIBIT and the ADS B INHIBIT switches must be placed in INHIBIT **and then** the ADS LOGIC A TIMER and ADS LOGIC B TIMER pushbuttons must be depressed.

ANSWER:

a. **open.**
The ADS A INHIBIT and the ADS B INHIBIT switches must be placed in INHIBIT.

FOILS:

b. ADS valve logic is satisfied and the valves are open.

c. ADS valve logic is satisfied and the valves are open.

d. Depressing the reset push buttons is not required.

REFERENCE: COR0021602, PR 2.2.1 Section 4.1, PR 2.4.4.1 Section 4.1.3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000.A4.07	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-29-02
18

Learning Objective: See Attached (As available)

Question Source:
Bank # 3268
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	3268	02	12/07/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022902001050D, COR0022902001080A, COR0022902001080H Standby Liquid Control System

Related Lessons
COR0022902 OPS STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001080H	Given a SLC component manipulation, predict and explain the changes in the following: Lights and alarms

Related References	
COR0022902	OPS STANDBY LIQUID CONTROL

Related Skills (K/A)	
211000.A4.07	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Lights and alarms 3.6 / 3.6

QUESTION: 23 3268 (1 point(s))

The plant is operating at 100% power when an ATWS occurs, you are given the order to inject SLC.

What indication that the Squib Valves operated correctly do you have on Panel 9-5?

a. Photohelic milliamp meters read zero.

b. White indicating lights are off.

c. Valve indicating lights are red.

d. SLC Pump Discharge pressure greater than Reactor pressure.

ANSWER: 23 3268

b. White indicating lights are off.

Explanation: When the SLC system pumps are started, the squib valves fire, setting off a charge that disrupts the continuity of the firing circuit. The way that this is indicated in the control room on panel 9-5 are the extinguishment of the two white lights for the squib valves.

Distractors:

a. The Photohelic milliamp meter reading zero is a good indication that the squib valves have been fired, but those meters are not available to the operator on Panel 9-5.

c. The only indication for the squib valves is the continuity lights; they do not have red and green position indication lights.

d. This indication would indicate that the SLC system pumps were running but not that the squib valves have operated correctly. Since these are positive displacement pumps they will develop a discharge pressure even if the relief valve (1400 psig setpoint) were the only flow path. The pressure instrument is upstream of the squib valve.

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

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-19-02
(Attach if not previously provided)
(including version/revision number) 20

Learning Objective: See Attached (As available)

Question Source: Bank # 24839
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	24839	01	12/07/2010		Licensed Operator	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Systems	The effect a loss of REC will have on the RWCU system

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060J	Given a specific REC malfunction, determine the effect on any of the following: RWCU system

Related References	
COR0021902	REACTOR EQUIPMENT COOLING

Related Skills (K/A)	
400000.A3.01	Ability to monitor automatic operations of the CCWS including: (CFR: 41.7 / 45.7) Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS 3.0 / 3.0

QUESTION: 24 24839

The Plant is operating at near rated power, when the following occurs:

- REC system supply header (DIV I, REC-PS-452A) experiences a low pressure of 59 psig for 1 minute.

What system will be affected FIRST?

a. The REC pumps will trip.

b. RWCU system will isolate.

c. Reactor Recirc Pumps will trip.

d. SW Quad temperatures will rise.

ANSWER: 24 24839

b. RWCU system will isolate.

Explanation:

With Div I header pressure below the isolation setpoint of 61.2 psig for more than 40 seconds the MO 700 goes closed. This isolates non-critical REC loop which causes a loss of cooling non-critical loads. Loss of cooling to the RWCU NRHX will result in RWCU high temperature and RWCU isolates on NRHX high temp.

Distractors:

a. This isolation will not cause any REC pump trips.

c. Reactor Recirc Lube Oil Heat exchangers loose cooling but it takes time for the oil to increase in temperature to the trip setpoint. The RWCU system will isolate first.

d. REC-MO-11 remains open so there is no affect on the quad fan cooling units.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001.K4.03	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR001-01-02
(Attach if not previously provided)
(including version/revision number) 34

Learning Objective: See Attached (As available)

Question Source: Bank # 1098
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	1098	01	09/20/2005		Licensed Operator	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010102, If breaker 1FA is closed before 1AF, which of the following describes the status of 4160V bus 1F AND its supply breakers

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001090B	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Circuit breaker automatic trips

Related References	
2.2.18	4160V Auxiliary Power Distribution System

Related Skills (K/A)	
262001.K4.03	Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Interlocks between automatic bus transfer and breakers (3.1/3.4)

QUESTION: 25 1098

Given the following conditions:

- A loss of power to 4160V Bus 1A occurred.
- 4160V Bus 1F is supplied from the Emergency Transformer.
- Power has been restored to 4160V bus 1A.
- ALL breaker control switches have been flagged to match actual breaker position.
- 4160V Bus 1F is to be transferred to bus 1A.

If breaker 1FA is closed before 1AF, which of the following describes the status of 4160V bus 1F and its supply breakers?

Bus 1F will…

a. deenergize THEN reenergize from the diesel.

b. remain energized BUT breaker 1FA will trip open.

c. deenergize THEN reenergize from the emergency transformer.

d. deenergize AND remain deenergized until manual operator action is taken.

ANSWER: 25 1098

a. deenergize THEN reenergize from the diesel.

Explanation:

A caution plate on Panel C states that breaker 1AF (1BG) must be closed prior to closing breaker 1FA (1GB). There is an interlock between these two breakers to prevent back feeding 4160V bus 1A (1B) from 4160V bus 1F (1G). This is accomplished by a trip circuit for breaker 1AF (1BG) which senses its control switch in Normal-After-Close <u>and</u> both 1AN (1BN) and 1AS (1BS) breakers open simultaneously.

Distractors:

b. The bus de-energizes 1AF trips open.

c. Breaker 1AS trips open

d. The diesel is not isolated from the bus and will re-energize the buss on a loss of power.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	239002.A2.03	
Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedures 2.4SRV, 2.1.4, 2.1.4.1, & 2.1.5
11, 128, 32, 64

Learning Objective: See Attached (As available)

Question Source:
Bank #
Modified Bank # 3397 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26		00	12/07/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320125A0A0400 CNS Abnormal Procedures Reactor Pressure Control

Related Lessons	
INT0320125	CNS Abnormal Procedures (RO) Reactor Pressure Control

Related Objectives	
INT0320125N0N0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
2.4SRV	Stuck Open Relief Valve

Related Skills (K/A)	
239002.A2.03	Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Stuck open SRV 4.1/ 4.2*

QUESTION: 26 (1 point(s))

Given the following conditions:

- Reactor power is 85%.
- Temperature down stream of Relief Valve MS-RV-71F is 310°F **AND** rising.
- Annunciator, RELIEF VALVE OPEN, is alarming.
- Annunciator, SUPPR POOL DIV I WATER HIGH TEMP, is alarming.
- Suppression Pool temperature is 97°F **AND** rising.
- Subsequent actions have been unsuccessful in closing MS-RV-71F.

What actions are now required?

a. Reduce power until feedwater flow is between 5.2 to 6.5 Mlbm/hr and then place the Mode Switch in SHUTDOWN.

b. Coordinate load reduction with Load Dispatcher. Ensure all IRM Range switches have been cycled. Ensure the Rx pressure and temperature recorders are in service.

c. Inform Ops Management, Outage Director, Load Dispatcher and the GMPO of the intent to rapidly shutdown plant. Have RE provide guidance for continuous rod insertion during shutdown. Have Chemistry/RP install SJAE off-gas rad monitor bug sources.

d. Lower core flow to $40x10^6$ lbs/hr. Transfer 4160V Buses 1A through 1D to the Startup Transformer, if time permits. Concurrently enter Mitigating Task Scram Actions, Reactor Power Control, Reactor Water Level Control, Reactor Pressure Control, Balance Of Plant Actions.

ANSWER: 26

d. Lower core flow to $40x10^6$ lbs/hr. Transfer 4160V Buses 1A through 1D to the Startup Transformer, if time permits. Concurrently enter Mitigating Task Scram Actions, Reactor Power Control, Reactor Water Level Control, Reactor Pressure Control, Balance Of Plant Actions.

Explanation:

Procedure 2.4SRV tells the Operator to enter procedure 2.1.5 Reactor Scram when the SRV cannot be closed.

Distractors:

a. Old steps from the rapid power reduction procedure.

b. First steps out of the current Normal Shutdown Procedure, 2.1.4.

c. First steps out of the current Rapid Shutdown Procedure, 2.1.4.1.

Modified: 3397

QUESTION: 3397

Given the following conditions:

- Reactor power is 85%.
- Temperature down stream of Relief Valve MS-RV-71F is 310°F **AND** rising.
- Annunciator, RELIEF VALVE OPEN, is alarming.
- Annunciator, SUPPR POOL DIV I WATER HIGH TEMP, is alarming.
- Suppression Pool temperature is 97 °F **AND** rising.
- Main generator load has lowered.
- Subsequent actions have been unsuccessful in closing MS-RV-71F.

Which of the below actions is now required?

a. Enter procedure 2.1.4, Normal Shutdown.

b. Enter procedure 2.1.4.1, Rapid Shutdown.

c. Enter procedure 2.1.5, Emergency Shutdown from Power.

d. Reduce power until feedwater flow is between 5.2 to 6.5 Mlbm/hr, **THEN** place the Mode Switch in SHUTDOWN.

ANSWER:

c. Enter procedure 2.1.5, Emergency Shutdown from Power.

REFERENCES: PR 2.4SRV

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000.K1.01	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson INT008-06-13
(Attach if not previously provided)
(including version/revision number) 14

Learning Objective: See Attached (As available)

Question Source: Bank # 1741
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 8
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	1741	01	12/08/2010	05/23/2010	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Systems	What is the purpose of the Residual Heat Removal Suppression Pool Spray mode of operation?

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001040B	Describe the interrelationship between the RHR system and the following: Suppression Pool
COR0022302001060L	Given an RHR control manipulation, predict and explain changes in the following: Containment parameters (pressure, temperature)

Related References	
COR0022302	Residual Heat Removal

Related Skills (K/A)	
230000.K1.01	Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Suppression pool (3.6/3.7)

QUESTION: 27 1741 (1 point(s))

How is the RHR System Suppression pool spray mode of operation initiated and what is its purpose?

a. manually initiated to reduce the Suppression Pool Temperature.

b. automatically initiated to reduce the Suppression Pool Temperature.

c. manually initiated to draw the non-condensables from the Drywell into the Torus.

d. automatically initiated draw the non-condensables from the Drywell into the Torus.

ANSWER: 27 1741

c. manually initiated to draw the non-condensables from the Drywell into the Torus.

Explanation:

From the EOP Flowchart 3A Lesson INT0080613 - Torus sprays are started between a torus pressure of 1.84 psig (high drywell pressure scram set-point) and 10 psig (Suppression Chamber Spray Initiation Pressure). Below 1.84 psig, normal methods of pressure control are to be employed. The actual setpoint of the pressure switches in the RHR logic prevent placing sprays in service until drywell pressure reaches ~ 2.5 psig.

The Suppression Chamber Spray Initiation Pressure, SCSIP, is defined to be the lowest torus pressure which can occur when 95% of the non-condensables (N_2) in the drywell have been transferred to the torus. This SCSIP is used to preclude chugging (the cyclic condensation of steam at the downcomer openings of the drywell vents).

Distractors:

a. it is manually initiated, however the purpose is the transfer the non-condensables to the torus. Suppression Pool Temperature will lower as a result, but it is not the purpose of Torus Sprays.

b. it is manually initiated, however the purpose is the transfer the non-condensables to the torus. Suppression Pool Temperature will lower as a result, but it is not the purpose of Torus Sprays.

a. it is manually initiated. There are no automatic initiation functions of this mode.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002.K2.02	
	Importance Rating	2.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-22-02, 26

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28		0	12/08/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Systems	What is the power supply to the “A” Rx Recirc MG-Set?

Related Lessons	
COR0022202	Reactor Recirculation

Related Objectives	
10A	Identify the power supplies to the following: a. RRMG set drive motors

Related References	
COR0022202	Reactor Recirculation

Related Skills (K/A)	
202002.K2.02	Knowledge of electrical power supplies to the following: Hydraulic power unit (2.6 / 2.6)

QUESTION: 28 (1 point(s))

What is the normal power supply to the “A” Reactor Recirculation Pump Motor Generator Set?

a. 4160 VAC Bus A.

b. 4160 VAC Bus B.

c. 4160 VAC Bus C.

d. 4160 VAC Bus D.

ANSWER: 28

c. 4160 VAC Bus C.

Explanation:

The Reactor Recirc Pump Motor Generator Sets can be supplied via two power supplies, the Normal Transformer and the Startup Transformer. One pump is normally aligned to the Normal Transformer during power operation, and typically it is the “A” RR Pump through breaker 1CN. The “B” RR Pump receives its power typically from the Startup transformer through breaker 1DS.

Distractors:

a. This bus supplies equipment like the Circ Water and Condensate Pumps, but not the Reactor Recirc Pumps.

b. This bus supplies equipment like the Circ Water and Condensate Pumps, but not the Reactor Recirc Pumps.

d. This bus supplies the “B” Reactor Recirc Pumps.

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

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-23-02
27

Learning Objective: See Attached (As available)

Question Source:
Bank #
Modified Bank # 1752 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29		00	12/09/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	2	Multiple Choice	

Topic Area	Description
Systems	COR0022302001080A, COR0022302001150B Residual Heat Removal System

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)
COR0022302001150B	Given plant conditions, determine if the following should occur: RHR pump start

Related References
COR0022302 RESIDUAL HEAT REMOVAL

Related Skills (K/A)	
219000.K3.01	Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE will have on following: (CFR: 41.7 / 45.4) Suppression pool temperature control (3.9 / 4.1)

QUESTION: 29 (1 point(s))

The plant is operating at 100% power with the HPCI Full Flow Surveillance in progress and “B” RHR pump in Suppression Pool Cooling, maintaining torus water temperature at 93°F, when the following happens:

- The “B” RHR Pump trips on an electrical fault.

The Operator places the Control Switch for the “D” RHR in the Start position. What will be the “D” RHR pump response and what affect will this have on Suppression Pool Temperature?

	“D” RHR Pump	**Torus water temperature**
a.	will not start	starts rising
b.	will not start	remains constant
c.	starts	remains constant
d.	starts	starts lowering

ANSWER: 29

c. starts remains constant

Explanation:

With the “B” and “D” pumps being in the same loop but powered off different power supplies, the electrical fault would only be experienced in the “B” Pump. The “D” pump would start right away and the valves would still be aligned at their same position before the “B” pump trip, so the “D” pump would take the place of the “B” pump delivering the same cooling reduction and would maintain Torus temperature the same.

Distractors:

a. “D” pump is not affected by the electrical fault on the “G” Buss.

b. “D” pump is not affected by the electrical fault on the “G” Buss. Therefore torus temperature would rise if the pump could not be started.

d. The “D” Pump starts but is at the same flow rate and cooling rate as the “B” pump, so the temperature should remain constant, not lower.

Modified 1752

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003.K4.08	
	Importance Rating	2.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-05-02
10

Learning Objective: See Attached (As available)

Question Source:
Bank # 2092
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	2092	00	08/12/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020502001020L, COR0020502001050H, COR0020502001120E Control Rod Drive Mechanisms

Related Lessons
COR0020502 CONTROL ROD DRIVE MECHANISM

Related Objectives	
COR0020502001020L	State the purpose of the following major CRDM components: Position Indicator Probe Thermocouple
COR0020502001050H	Given the CRDM design features and/or interlocks that provide for the following: monitoring CRDM temperatures
COR0020502001120E	Determine the interrelationships between the CRDMs and the following: CRDM Temperature Monitor

Related References	
COR0020502	CONTROL ROD DRIVE MECHANISM

Related Skills (K/A)	
201003.K4.08	Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: Monitoring CRD mechanism temperature (2.6 / 2.7)

QUESTION: 30 2092 (1 point(s))

What is the function of the Position Indicator Probe thermocouple located in each Control Rod Drive Mechanism?

The Position Indicator Probe thermocouple provides…

a. remote indication of CRDM operating temperatures.

b. the signal for the high CRDM temperature annunciator on panel 9-5.

c. temperature compensation of the reed switch output current signal.

d. post-LOCA (loss of coolant accident) temperature indication for accident analysis.

ANSWER: 30 2092

a. remote indication of CRDM operating temperatures.

Explanation:

In accordance with the Student Text COR0020502, the Rod Position Information System (RPIS) provides for the following:

- The position indicating probe (PIP) supplies the input to the Rod Position Information System.
- The thermocouple located in the PIP gives indication of CRDM temperature.

Distractors:

b. The actual signal comes from the PMIS Computer, not the thermocouples.

c. There is no temperature compensation for the output current for the reed switches.

d. The temperature probes are not credited for any post LOCA analysis.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	233000.K5.07	
	Importance Rating	2.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) USAR X Section 5 / Procedure 2.4FPC
(Attach if not previously provided)
(including version/revision number) 2/21/07 22

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31		00	12/15/2010		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
Systems	FP Cooling - What is HIGHEST temperature expected and what are the implications?

Related Lessons
COR0010602 FUEL POOL COOLING

Related Objectives	
COR0010602001100D	Briefly describe the following concepts as they apply to FPC: Heat loading

Related References	
USAR	X Section 5
PROC	2.4FPC.

Related Skills (K/A)	
233000.K5.07	Knowledge of the operational implications of the following concepts as they apply to FUEL POOL COOLING AND CLEAN-UP: (CFR: 41.5 / 45.3) Maximum (abnormal) heat 102d load (2.5 / 2.8)

QUESTION: 31

What is HIGHEST fuel pool temperature expected under design heat load and what are the implications for exceeding that temperature?

a. 125 °F; RHR sub-system would have to be placed in service to support cooling.

b. 125 °F ; RWCU system would have to be placed in service to support cooling.

c. 150 °F ; RHR sub-system would have to be placed in service to support cooling.

d. 150 °F ; RWCU system would have to be placed in service to support cooling.

ANSWER: 31

c. 150 °F ; RHR sub-system would have to be placed in service to support cooling.

Explanation:

USAR – X Section 5: During normal refueling operations, the maximum expected spent fuel pool temperature of 150°F results from the decay heat of the full core load of fuel at the end of the fuel cycle plus the remaining decay heat of the spent fuel discharged at previous refuelings. Prior to the spent fuel pool reaching this temperature, the Residual Heat Removal system is manually aligned to operate in conjunction with the fuel pool cooling and demineralizer system to reduce the spent fuel pool temperature and maintain it at or below 150°F.

Distractors:

a. This is the entry condition for a loss of fuel pool cooling. But RHR is the eventual supplemental cooling system.

b. This is the entry condition for a loss of fuel pool cooling. And RWCU is not the eventual supplemental cooling system.

d. This is the correct temperature, but RWCU is not the supplemental cooling system.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000.K6.01	
	Importance Rating	2.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.3NBPP 11; Lesson COR002-20-02 20

Learning Objective: See Attached (As available)

Question Source:
Bank #
Modified Bank # 2107 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32		00	12/15/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	

Topic Area	Description
Systems	What does RPIS indication look like on Panel 9-5 on a loss of NBPP?

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001100D	State the electrical power supplies to the following: RPIS
COR0022002001050C	Predict the consequences the following would have on the RMCS and/or RPIS: RPIS failure

Related References	
COR0022002	OPS REACTOR MANUAL CONTROL SYSTEM
PROC.	5 .3NBPP

Related Skills (K/A)	
214000.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM: (CFR: 41.7 / 45.7) A.C. electrical power (2.5 / 2.6)

QUESTION: 32 (1 point(s))

The Plant is operating at approximately 75% with Control Rod 26-27 selected.

- There is a loss of the No Break Power Panel (NBPP).

What is the expected condition of the Control Rod Position Indication System and the indication displayed in the four-rod display on the vertical Panel 9-5.

	Full Core Display	4 Rod Display
a.	No Lights	No position
b.	Lights	No Position
c.	Lights	Position Indicated
d.	No Lights	Position Indication

ANSWER: 32

a. No Lights No position

Explanation:

The No Break Power System powers all indications for Control Rod Positions. Both the red and green lights on the full core display and the numeric indications located on the four rod display.

From 5.3NBPP - A loss of NBPP causes the following

Distractors:

b. There are no red and green lights on the Full Core Display.

c. There are no red and green lights on the Full Core Display.

d. There are no numerals displayed in the four rod display.

Modified: 2107

QUESTION: 2107

As the reactor operator, you note a loss of control rod position indication on the four-rod display and suspect a failure of the Rod Position and Information System's power supply. What electrical distribution panel should you instruct your station operator to check?

a. Critical Power Panel

b. 125 VDC Distribution Panel

c. No Break Power Panel

d. Reactor Protection System Power Panel

ANSWER:

c. No Break Power Panel

REFERENCE: Reactor Manual Control System Text

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000.A1.06	
	Importance Rating	3.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-09-02
15

Learning Objective: See Attached (As available)

Question Source:
Bank # 2399
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	2399	02	12/16/2010		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
Systems	What happens when a bypass opens at 70% power?

Related Lessons
COr0020702001080B Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001070D	Given a specific DEH Control system malfunction, determine the effect on any of the following: Main turbine steam flow

Related References
COR0020702 Digital Electro-Hydraulic Control

Related Skills (K/A)	
241000.A1.06	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: (CFR: 41.5 / 45.5) Main turbine steam flow (3.2 / 3.2)

QUESTION: 33 2399 (1 point(s))

The Plant is operating at 70% power with DEH in mode IV, when the following event occurs:

- One bypass valve partially opens

What is the expected response on Steam Flow and overall plant?

a. Steam flow increase, steam header pressure decrease, Group I isolation on low Rx. Pressure.

b. Steam flow decrease, steam header pressure increase, scram on Rx. high pressure.

c. Steam flow increase, steam header pressure maintained by governor valves, generator output increase.

d. Steam flow and steam header pressure remain constant, generator output decrease.

ANSWER: 33 2399

d. Steam flow and steam header pressure remain constant, generator output decrease.

Explanation:

The LOAD CONTROL mode is entered when the first of either generator output breakers is closed (3310 or 3312). DEH is normally in this mode for only a brief period of time as generator output is automatically ramped up until the turbine BPVs close. In this case, the bypass valve opens thus transferring DEH back into MODE 3. The turbine BPVs which are controlling pressure, if they open, throttle pressure lowers and the Governor Valves close down to maintain throttle pressure. Total Steam Flow (that is going through the Governor valves and the bypass valve) remain constant, however generator load decreases as the governor valves close.

Distractors:

a. Total steam flow remains constant as the governor valve close down to compensate for the open bypass valve. Therefore maintaining Reactor Pressure constant.

b. Total steam flow remains constant as the governor valve close down to compensate for the open bypass valve. Therefore maintaining Reactor Pressure constant.

c. Total steam flow remains constant as the governor valve close down to compensate for the open bypass valve. Therefore maintaining Reactor Pressure constant.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215001.A2.07	
	Importance Rating	3.4	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-31-02
(Attach if not previously provided)
(including version/revision number) 14

Learning Objective: See Attached (As available)

Question Source: Bank # 21332
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	21332	00	02/08/2005	05/23/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	5	Multiple Choice	

Topic Area	Description
Systems	COR0023102, TIP Loss of Power

Related Lessons
COR0023102 OPS TRAVERSING IN-CORE PROBE

Related Objectives	
COR0023102001140B	Predict the consequences of the following on the TIP system: A.C. Electrical power failure

Related References	
4.1.4	Traversing In-Core Probe System

Related Skills (K/A)	
215001.A2.07	Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct...: (CFR: 41.5 / 45.6) ?Failure to retract during accident conditions: Mark-I&II(Not-BWR1) (3.4/3.7)

QUESTION: 34 21332 (1 point(s))

The plant is operating at power with TIP "A" near the core top limit when annunciator C-4/F-6, CRIT INST & CONT PNL CPP LOSS OF VOLT alarms. While investigating the cause of the loss of CPP, a reactor scram occurs due to high drywell pressure. Drywell pressure is 5 psig and rising.

What TIP action(s) is/are required in order to isolate the TIP tube? (Choose the answer that contains ONLY required TIP actions).

a. Place TIP "A" MANUAL switch to REV only.

b. Place the key lock switch for TIP "A" shear valve to FIRE.

c. Place TIP "A" MAN. VALVE CONTROL switch to CLOSED only.

d. Place TIP "A" MANUAL switch to REV and Place MAN. VALVE CONTROL switch to CLOSED.

ANSWER: 34 21332

b. Place the key lock switch for TIP "A" shear valve to FIRE.

The loss of CPP results in the loss of TIP instrumentation and controls. Since an isolation is required and the TIP cannot be withdrawn from the core the shear valve should be fired.

Distractors:

a, c, d Valve Control Monitor and the four drive control units are fed from CPP. Loss of this power supply will cause the drive mechanism to think that the tip is in its shield. The TIP will stop where ever it is and the ball valve will attempt to close.

Psychometric Review: GPJ

2004 Biennial Exam 04LORRO01 Question # 21055

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000.A3.05	
	Importance Rating	3.0	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-02-02, 30

Learning Objective: See Attached (As available)

Question Source:
- Bank #
- Modified Bank # 3963 (Note changes or attach parent)
- New

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis H

10 CFR 55 Content:
- 55.41 7
- 55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35		00	12/16/2010		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020202, Which Cond & Booster pumps are running after power loss

Related Lessons
COR0020202 OPS CONDENSATE AND FEEDWATER

Related Objectives
COR0020202001080E Predict the consequences a malfunction of the following would have on the Condensate and Feedwater system: AC Power

Related References	
COR0020202	OPS CONDENSATE AND FEEDWATER

Related Skills (K/A)
256000.A3.05 Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: (CFR: 41.7 / 45.7) Lights and alarms (3.0 / 2.9)

QUESTION: 35 (1 point(s))

The plant is at 100% power when a LOSS of both the Normal and Startup Transformers occurs.

What do the Condensate Pump indications show?

Condensate Pump…

	Amps Meters	Indicating lights
a.	≈ 80 Amps	Green Light off, Red light on
b.	≈ 80 Amps	Green Light on, Red light off
c.	0 Amps	Green Light off, Red light on
d.	0 Amps	Green Light off, Red light off

ANSWER: 35

c. 0 Amps Green Light off, Red light on

Explanation:

On a loss of 4160 Volt buses feeding the Condensate Pumps, the pumps will stop operating, however they do not have an under voltage trip so the red breaker closed light will remain on and the current will be 0 amps.

Distractors:

a. The 4160 Bus that feeds the Condensate pumps will be de-energized so there should be no current read on the Pump Amp Meters.

b. The 4160 Bus that feeds the Condensate pumps will be de-energized so there should be no current read on the Pump Amp Meters.

d. The 4160 Bus that feeds the Condensate pumps will be de-energized but there are no under voltage trips on the breakers feeding them, so the red light should remain on.

Modified - 3963

QUESTION: 3963

The plant is at 75% power when a LOSS of the Normal Station Transformer occurs.

- EXCEPT for Bus 1B, ALL 4160 VAC buses transfer successfully.
- Breaker 1BS fails to close automatically.
- One (1) second later, breaker 1BS is closed by the BOP operator AND energizes bus 1B.

What is the state (running or stopped) of the Condensate (COND) and Condensate Booster (CB) pumps powered from Bus 1B just after Bus 1B is energized?

	COND PUMP	CB PUMP
a.	RUNNING	RUNNING
b.	RUNNING	STOPPED
c.	STOPPED	RUNNING
d.	STOPPED	STOPPED

ANSWER:

a. RUNNING RUNNING

EXPLANATION OF ANSWER: a. Correct. CP and CBP breakers do not have UV protection and will remain closed upon loss of power. Pumps will coast down during the power loss and re-start when the bus is re-energized.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	291001.A4.01	
	Importance Rating	3.3	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Procedure 2.3_R-2
(Attach if not previously provided)
(including version/revision number) 14

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	7
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36		00	01/11/2011		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	Deviation from normal Secondary Containment pressure

Related Lessons
COR0020302 OPS CONTAINMENT

Related Objectives

Related References
Procedure 2.3_R-2

Related Skills (K/A)	
290001.A4.01	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Reactor building differential pressure: Plant-Specific (3.3 / 3.4)

QUESTION: 36

The Reactor is operating at 100% power when the Reactor Operator notes that Reactor Building differential pressure is -0.30 inches of water.

Is this pressure better or worse than the normal pressure and what is the trip setpoint for the reactor building fans? (Neglect the time delay)

a. better; - 0.15 inches of water.

b. better; + 0.15 inches of water.

c. worse; - 0.15 inches of water.

d. worse; + 0.15 inches of water.

ANSWER: 36

a. better; - 0.15 inches of water.

Explanation:

Normal differential pressure for the reactor building is – 0.25 inches of water. – 0.30 inches is more negative and therefore a better pressure to maintain. The trip setpoint for the Reactor Building fans is – 0.15 inches of water in accordance with annunciator procedure 2.3_R-2, window R-2/A-4. SETPOINT (4703) -0.15" wg. Or -0.45 inches of water.

Distractors:

b. even though the pressure is better, the fans trip on a negative pressure of 0.15 not a positive pressure.

c. a slightly more negative pressure is better than a more positive pressure.

d. a slightly more negative pressure is better than a more positive pressure, and the trip setpoint is not +0.15.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	2.2.22	
	Importance Rating	4.0	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) T.S. 3.7.4
(Attach if not previously provided)
(including version/revision number) Amendment 230

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37		00	12/29/2010		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	1	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070508, CNS Tech Spec 3.7, Plant Systems

Related Lessons
INT0070508 OPS Tech. Specs. 3.7, Plant Systems

Related Objectives	
INT00705080010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.7 LCO, determine the ACTIONS that are required.

Related References	
3.7.4	Control room emergency filter (CREF) system

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

QUESTION: 37

The plant is in MODE 1 at 100% power for 265 days when the following is discovered:

- The Control Room Envelope (CRE) has been breeched when a door contained within that boundary came off its hinges and fell to the floor.

What actions are required by Technical Specifications?

a. Enter LCO 3.0.3 immediately.

b. Immediately initiate actions to implement mitigating actions.

c. Be in MODE 3 (Hot Shutdown) in 12 hours and be in MODE 4 (Cold Shutdown) in 36 hours.

d. Immediately initiate actions to suspend movement of irradiated fuel assemblies in secondary containment.

ANSWER: 37

b. Immediately initiate actions to implement mitigating actions.

Explanation:

Tech Spec 3.7.4 Control Room Emergency Filter (CREF) System is applicable in MODES 1, 2, and 3 and will be declared inoperable if the Control Room Envelope (CRE) is breeched. That requires Condition B to be entered and Required Action B.1 to be initiated immediately.

Distractors:

a. This would only be required if there were no Required Actions for this condition.

c. This would only be required if the Required Actions and Associated Completion Times were not complied with.

d. There are no fuel assemblies being moved in secondary containment so this is not required.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000.K3.29	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-15-02 & COR002-22-02
(Attach if not previously provided)
(including version/revision number) 24 26

Learning Objective: See Attached (As available)

Question Source:			
	Bank #	1063	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	7
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	1063	00	06/22/1999		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

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
COR002-02-22	Nuclear Boiler Instrumentation

Related Skills (K/A)	
202002.K3.01	Knowledge of the effect that a loss or malfunction of the RECIRCULATION FLOW CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4) Core flow (3.5/3.5)
216000.K3.29	Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following: (CFR: 41.7 / 45.4) K3.29 Jet pump flow monitoring: Plant-Specific (3.1 / 3.2)

QUESTION: 38 1063 (1 point(s))

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 are the expected values for indicated Total Core Flow as indicated on Panel 9-5 Recorder NBI-FR/DPR-95 **AND** what is Actual Core Flow?

	Indicated Total Core Flow	**Actual Core Flow**
a.	37 Mlbm/hr	33 Mlbm/hr
b.	37 Mlbm/hr	37 Mlbm/hr
c.	33 Mlbm/hr	33 Mlbm/hr
d.	33 Mlbm/hr	37 Mlbm/hr

ANSWER: 38 1063

a. 37 Mlbm/hr 33 Mlbm/hr

Explanation:

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.

Distractors:

b. 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).

c. 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).

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).

COR002-22-02, PR 2.2.68.1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028.EK1.01	
	Importance Rating	3.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) PSTGs
(Attach if not previously provided)
(including version/revision number)

Learning Objective: See Attached (As available)

Question Source:			
	Bank #	24832	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	14
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	24832	00	07/18/2009		Licensed Operator	RO: SRO: NLO:	Y Y Y

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Integrated Plant	EOP 3A, why monitor DW temperature

Related Lessons
COR0021502 OPS NUCLEAR BOILER INSTRUMENTATION 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.
COR0021502001040F	Briefly describe the following concepts as they apply to NBI: Elevated containment temperature effects on level indication
COR0021502001050G	Predict the consequences of the following items on the NBI: Elevated containment temperature

Related References
NONE

Related Skills (K/A)	
295028.EK1.01	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: CFR: 41.8 to 41.10) Reactor water level measurement (3.5/3.7)

QUESTION: 39 24832 (1 point(s))

Which one of the following is a reason why Drywell Temperature is monitored and controlled by EOP-3A, PRIMARY CONTAINMENT CONTROL?

a. Ensure NPSH limits for ECCS pumps are not exceeded.

b. Verify proper operation of the Drywell Hydrogen detectors.

c. Prevent or minimize inaccurate indications of RPV pressure instruments.

d. Prevent or minimize inaccurate indications of RPV water level instruments.

ANSWER: 39 24832

d. Prevent or minimize inaccurate indications of RPV water level instruments.

Explanation:

In accordance with the EPGs and the EOP Bases, RPV water level indications may be unreliable or must be considered invalid due to the effects of increased Drywell temperatures.

Distractors:

a. NPSH limit for ECCS pumps is ensured by Torus minimum level.

b. Drywell temperature control is not a reason operation of the Hydrogen detectors is verified.

c. Torus pool temperature is monitored, not air temperature.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024.EK1.01	
	Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson INT008-06-13 ; INT008-06-18
(Attach if not previously provided)
(including version/revision number) 14 18

Learning Objective: See Attached (As available)

Question Source: Bank # 19220
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	19220	01	07/28/2003		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080618, Basis for PSP ED

Related Lessons	
INT0080613	OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.
INT0080613001040C	State the basis for primary containment control actions as they apply to the following: Graphs reference on Flowchart 3A
INT00806180010200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Related References	
INT0080613	OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Skills (K/A)	
295024.EK1.01	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) Drywell integrity: Plant-Specific (4.1/ 4.2*)

QUESTION: 40 19220 (1 point(s))

Why is the RPV Emergency Depressurized if Pressure Suppression Pressure (EOP Graph 10) is exceeded?

a. Failure of primary containment may occur if drywell sprays are initiated.

b. Failure of primary containment may occur if a primary system rupture develops.

c. Failure of SRV Tailpipes may occur due to steam bypassing the suppression pool.

d. Failure of SRV Tailpipes may occur due to inadequate differential pressure across the balancing disc.

ANSWER: 40 19220

b. Failure of primary containment may occur if a primary system rupture develops.

Explanation:
INT008-06-13
If primary containment water level rises above 16.5 ft, a return flowpath for non-condensibles from the torus to the drywell through the vacuum breakers may be prevented. Drywell sprays must therefore be stopped. If adequate core cooling cannot be assured (unless combustible gas concentrations exist in the primary containment), drywell sprays must be stopped.

PC/P-4 - If torus or drywell sprays could not be started or if their operation was not effective in reducing primary containment pressure, the RPV is depressurized when torus pressure cannot be maintained below the Pressure Suppression Pressure, PSP. The Alarm Statement following step PC/P-4 is needed to alert the CRS that an override in Flowchart 1A (or Flowchart 6A/7A for failure-to-scram events) must be reviewed. RPV depressurization minimizes further release of energy from the RPV to the primary containment. See INT008-06-18, GRAP10 for discussion of the PSP.

INT008-06-18
Pressure Suppression Pressure (GRAP10)
Definition - The Pressure Suppression Pressure, PSP, is the highest torus pressure which can be maintained without steam in the torus airspace.

Use - The PSP is a function of primary containment water level. It is utilized in the EOPs to ensure that pressure suppression capability sufficient to accommodate emergency depressurization is maintained while the RPV is at pressure. It is utilized in the SAGs to ensure that pressure suppression capability sufficient to accommodate a low pressure release of core debris is maintained when RPV breach by core debris is anticipated.

Flowchart 3A step PC/P-4 requires emergency RPV depressurization when torus pressure cannot be maintained within the PSP.

If RPV water level is below the bottom of active fuel and RPV injection flow is below MDRIR, SAG 2 Strategy E is performed based on the ability to maintain parameters within the PSP; Strategy F, if unable to maintain parameter within PSP.

Distractors:

a. PSP is not based on drywell spray restrictions (DWSIL is met).

c. SRVTPLL is not based on steam bypassing the suppression pool.

d. SRVTPLL is not based on d/p on balancing disc (PCPL is met).

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

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) USAR XIV
(Attach if not previously provided)
(including version/revision number) 2/5/10

Learning Objective: See Attached (As available)

Question Source: Bank # 19212
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam CNS 2002

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

10 CFR 55 Content 55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	19212	01	03/17/2007		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070510, CNS Tech. Spec. 3.9, Refueling Operations (2002 NRC Exam)

Related Lessons	
INT0070510	OPS CNS Tech. Spec. 3.9, Refueling Operations

Related Objectives	
INT00705100010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.9 Specification.

Related References
NONE

Related Skills (K/A)	
295023.AK1.03	Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: (CFR: 41.8 to 41.10) Inadvertent criticality. (3.7/4.0)

QUESTION: 41 19212 (1 point(s))

What is the basis for the refueling interlocks associated with the Reactor Mode Switch in the REFUEL position?

a. To prevent criticality during refueling by ensuring that fuel assemblies are not loaded into the core unless all control rods are fully inserted.

b. To prevent control rod/fuel assembly configurations during refueling resulting in fuel enthalpy above 280 cal/gm by preventing multiple control rod withdrawal.

c. To ensure that fuel assembly loading sequence and configurations are restricted to maintain an adequate shutdown margin by monitoring control rod position and refueling grapple load status.

d. To ensure that radioactivity releases as a result of a refueling accident are maintained below a small fraction of the 10CFR100 limits by preventing more than one hoist from being loaded at a given time.

ANSWER: 41 19212

a. To prevent criticality during refueling by ensuring that fuel assemblies are not loaded into the core unless all control rods are fully inserted.

REFERENCE: Tech Spec 3.9.1 Bases

Foils:

b. 280 cal/gm is a RWM bases (N/A during refueling). The refueling interlocks place no restrictions on fuel assembly configuration.

c. The refueling interlocks do not ensure an adequate shutdown margin.

d. The refueling interlocks do not prevent loading of multiple hoists at the same time and the loading of the hoists is not related to the 10CFR100 release rates.

THIS QUESTION WAS USED ON THE 2002 CNS NRC EXAMINATION AS RECORD # 17887.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018.AK2.01	
	Importance Rating	3.3	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Emergency Procedure 5.2REC
(Attach if not previously provided)
(including version/revision number) 12

Learning Objective: See Attached (As available)

Question Source: Bank # 25186
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	25186	02	03/29/2010	05/23/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	Coolable REC loads following an REC system low pressure

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001050A	Briefly describe the following concepts as they apply to REC: Leak or lowering system pressure during accident and transient conditions
COR0021902001060A	Given a specific REC malfunction, determine the effect on any of the following: REC header pressure

Related References	
2.2.65	Reactor Equipment Cooling Water System
5.2REC	Loss Of REC

Related Skills (K/A)	
295018.AK2.01	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8) System loads (3.3/3.4)

QUESTION: 42 25186 (1 point(s))

A Loss of Offsite Power (LOOP) occurred. #1 and #2 Emergency Diesel Generators automatically start and power the critical busses. Two (2) minutes later, REC pressure has stabilized at 50 psig. Access to the Reactor Building is prohibited by radiation levels.

Which of the following loads can be provided long term cooling using the REC system?

a. Augmented Off Gas (AOG)

b. "A" Control Rod Drive pump

c. "B" Drywell Fan Coil Unit

d. HPCI Room Fan Coil Unit

ANSWER: 42 25186

d. HPCI Room Fan Coil Unit

Explanation:

With an isolation signal present REC-MO-702MV can be reopened, however, the REC-MO-712 and 713 will auto close on the low pressure and cannot be overridden. This will isolate REC to the non-critical loops/components.

Distractors:

a. AOG is isolated

b. This would require a manual operation.

c. The Drywell is isolated

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005.AK2.01	
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) USAR XIV – 5
(Attach if not previously provided)
(including version/revision number) 9/19/00

Learning Objective: See Attached (As available)

Question Source:			
	Bank #	24799	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	7
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	24799	00	07/15/2009		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	

Topic Area	Description
Updated Safety Analysis Report	M. Gen output breakers open, cause of Rx Scram

Related Lessons	
INT0060119	Anticipated Operational Transients and Special Events
COR0022102	REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001100E	Describe the interrelationship between the RPS and the following: DEH
INT00601190010200	Given a Anticipated Operational Transient that is analyzed in the CNS USAR, select an action or actions that will terminate the transient.
COR0022102001040M	Describe the RPS design features and/or interlocks that provide for the following: Related system inputs to RPS

Related References
NONE

Related Skills (K/A)	
295005.AK2.01	Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: (CFR: 41.7 / 45.8) RPS. (3.8/3.9)

QUESTION: 43 24799 (1 point(s))

The Plant is operating at near rated power, with the following conditions:

- PCB-3312 is Tagged open due to maintenance in the 345 Switch yard
- PCB-3310 Auto Reclosure Switch is in OFF

Then the following sequence of events occur:

- A line fault causes PCB 3310 to Open
- Automatic Reactor Scram is initiated

What is the cause of the Reactor Scram?

a. High Reactor Pressure Scram

b. HI APRM Scram (Neutron monitoring Scram)

c. Reactor Scram due Turbine Stop Valve Closure.

d. Reactor Scram due to Turbine Control valve fast closure.

ANSWER: 43 24799

d. Reactor Scram due to Turbine Control valve fast closure.

Explanation:

As shown in USAR Figure XIV-5-1, as soon as turbine control valve fast closure is sensed, a scram is initiated. This occurs in advance of the high neutron flux and high RPV pressure scram signals thereby limiting the peak neutron flux to about 180 percent of rated. The average surface heat flux reaches a peak of about 115 percent of rated. The small increase in average surface heat flux coupled with the slight increase in core flow ensures that nucleate boiling is maintained throughout the transient. The Reactor Scram due to Turbine Stop valve closure is a separate analyzed transient. The stop valve closure assumes a direct turbine trip signal not a load rejection.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019.AK2.01	
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Ann Procedure 2.3_9-5-2
(Attach if not previously provided)
(including version/revision number) 28

Learning Objective: See Attached (As available)

Question Source:		
Bank #	3996	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content	
55.41	7
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	3996	02	10/14/2005		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0020502001110A Control Rod Drive Mechanisms

Related Lessons
COR0020502 CONTROL ROD DRIVE MECHANISM COR0020402 OPS CONTROL ROD DRIVE HYDRAULICS

Related Objectives	
COR0020502001110A	Given a specific CRDM malfunction, determine the effect on any of the following: Rod Movement
COR0020402001110A	Predict the consequences a malfunction of the following would have on the CRDH systems: Loss of plant air system
COR0020402001130F	Describe the interrelationships between the Control Rod Drive Hydraulic System (CRDH) and the following: Plant Air Systems

Related References	
2.3_9-5-2	Panel 9-5 - Annuciator 9-5-2

Related Skills (K/A)	
295019.AK2.01	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8) CRD hydraulics (3.8/3.9)

QUESTION: 44 3996 (1 point(s))

Given the following conditions;

- The plant is at 100% power.
- IA-PRV-PRV614, CRD Reliable Air Supply regulator fails so that output pressure slowly fails to 0 psig.
- Annunciator 9-5-1/F-5, SCRAM VALVE PILOT AIR LOW PRESSURE is received.
- NO operator action is taken.

Which of the following statements describes the status of the Control Rod Drive Hydraulics System for these conditions?

a. ALL Control Rods will scram randomly.
Normal rod insertion AND withdrawal will NOT be available

b. ALL Control Rods will scram randomly.
Normal rod insertion AND withdrawal will remain available

c. Control Rod Scram capability will be lost.
Normal rod insertion AND withdrawal will remain available

d. Control Rod Scram capability will be lost.
Normal rod insertion AND withdrawal will NOT be available

ANSWER: 44 3996

a. ALL Control Rods will scram randomly.
Normal rod insertion AND withdrawal will NOT be available.

Explanation:
PRV614 supplies air to the Scram Pilot Air Header and Flow Control Valves. The Flow Control Valves fail closed on loss of air causing a loss of normal drive capability. A loss of air to the Scram Pilot Air Header will cause the Scram Valves to drift open and rods will drift in as each Scram Valve starts to open.

Distractors:
b, The Flow Control Valves fail closed on loss of air causing in a loss of normal drive capability.

c. The Flow Control Valves fail closed on loss of air causing in a loss of normal drive capability.

d. A loss of air causes Scram Valves to open and rods to insert.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295021.AK3.01	
Importance Rating	3.3	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-23-02 Procedure 2.4SDC

Learning Objective: See Attached (As available)

Question Source:
Bank # 2751
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43 ______

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	2751	00	08/25/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022302001090D Residual Heat Removal System

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001090D	Explain the significance of the following as they apply to a loss of Shutdown Cooling: Natural circulation

Related References	
COR0022302	Residual Heat Removal
2.4SDC	Loss of Shutdown Cooling

Related Skills (K/A)	
295021.AK1.04	Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.8 to 41.10) Natural circulation. (3.6/3.7)
295021.AK3.01	Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5 / 45.6) Raising reactor water level (3.3 / 3.4)

QUESTION: 45 2751 (1 point(s))

With a loss of shutdown cooling, vessel water level is to be maintained > 48 inches.

What is the significance of this level?

a. To ensure adequate core cooling

b. To ensure natural circulation will occur.

c. To maintain required NPSH for the jet pumps.

d. To maintain adequate NPSH for the Reactor Recirculation pumps.

ANSWER: 45 2751

b. To ensure natural circulation will occur.

Explanation:

From System Lesson COR002-23-02 Natural Circulation - During plant cooldown, the reactor water level is maintained ≥ 48 inches to ensure natural circulation will occur should forced circulation be lost.

From Abnormal Procedure 2.4SDC - Contingency actions for complete loss of SDC:

- Commence monitoring plant heatup rate per Procedure 6 .RCS.601.
  NOTE: The Preferred level indication is NBI-LI-86, SHUTDOWN LVL.
  RFC-LI-94A,
- RFC-LI-94B, or RFC-LI-94C, RX NR LEVEL, may indicate up to 9" higher than actual during cold conditions.
- Control RPV level > 48" to aid in thermal convection flow.
- Monitor following temperatures and pressures frequently and log in Control Room Log every 4 hours:

Distractors:

a. Adequate Core Cooling is assured by submergence in this situation.

c. The Jet Pumps are not being used during a loss of Shutdown Cooling.

d. The Reactor Recirc Pumps are not being used during a loss of Shutdown Cooling.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003.AK3.06	
	Importance Rating	3.7	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-03-02
(Attach if not previously provided)
(including version/revision number) 27

Learning Objective: See Attached (As available)

Question Source:		
Bank #	25257	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content	
55.41	5
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	25257	00	04/14/2010	05/23/2010	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	3	Multiple Choice	

Topic Area	Description
Systems	Using panel indications discern RPS power condition.

Related Lessons
COR0020302 OPS CONTAINMENT COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0020302001060M	Describe the interrelationship between PCIS and the following: AC Distribution
COR0020302001200A	Briefly explain the reason for the following: Containment isolation on partial or complete loss of AC power
COR0020302001210B	Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations.
COR0022102001040L	Describe the RPS design features and/or interlocks that provide for the following: Under/over voltage and frequency protection
COR0022102001050C	Briefly describe the following concepts as they apply to RPS: EPA operation

Related References
NONE

Related Skills (K/A)	
295003.AK3.06	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.5 / 45.6) Containment isolation. (3.7/3.7)

QUESTION: 46 25257 (1 point(s))

Using the photograph below, of the 9-5 PCIS Group Isolation Indications.

What condition or malfunction caused the indications shown above.

Power has been lost or interrupted to…

a. only "A" RPS power panel.

b. only "B" RPS power panel.

c. both "A" and "B" RPS power panels.

d. neither "A" nor "B" RPS power panels.

ANSWER: 46 25257

a. only "A" RPS power panel.

Explanation:

On a loss of "A" RPS panel, a full Group 3 isolation will result due to the loss of the RWCU NRHX temperature switch relay. This relay is in both PCIS divisions. The full PCIS Group 6 is due to losing power to a Group 2 relay that is seen by the Gp 6 isolation circuit.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295037.EK3.03	
Importance Rating	4.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): PSTG Appendix B
3

Learning Objective: See Attached (As available)

Question Source:

Bank #	14478	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	H

10 CFR 55 Content

55.41	5
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	14478	02	12/12/2008	05/23/2010	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080610, How far to lower water level and why

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

Related Objectives	
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References

Related Skills (K/A)	
295037.EK3.03	Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.5 / 45.6) Lowering reactor water level (4.1*/4.5*)

QUESTION: 47 14478 (1 point(s))

The plant has experienced ATWS conditions with the following indications

• The feedwater system	in operation
• Reactor power	4% (stable)
• Reactor Pressure	935 psig (stable)
• Reactor water level	+13"(NR) (stable)
• Average torus water temperature	87ºF (stable)

How long must RPV injection be stopped and prevented and why?

Stop and prevent injection until…

a. reactor water level is less than - 60" (corrected FZ) to ensure incoming feedwater is heated.

b. reactor water level is less than - 60" (corrected FZ) to prevent the uncontrolled injection of large amounts of cold unborated water at low core flows.

c. the reactor is subcritical below the heating range to ensure reactor power is below the threshold for thermal hydraulic instabilities.

d. the reactor is subcritical below the heating range to ensure required feedwater injection rates reduce the chance of uncontrolled injection of large amounts of cold unborated water.

ANSWER: 47 14478

a. reactor water level is less than - 60" (corrected FZ) to ensure incoming feedwater is heated.

Explanation:

Reactor water level at - 60"(FZ) is below the level of the feedwater spargers which allows the incoming feedwater to be heated by the steam in downcomer.

Distractors:

b. is incorrect. Because the reason that level is lowered in this case is to ensure heating of the incoming feedwater. Stopping and preventing injection prior to emergency depress ensures no uncontrolled injection of relatively cold water.
c. is incorrect. Because water level need only be lowered to - 60"(FZ) even if reactor power remains elevated because 3 of the 4 level power conditions do not exist.
d. is incorrect. Because water level need only be lowered to - 60"(FZ) even if reactor power remains elevated because 3 of the 4 level power conditions do not exist.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000.AA1.01	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.3GRID Lesson COR001-13-01

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48		00	01/12/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	5	Multiple Choice	

Topic Area	Description
None	INT0320131, CNS Abnormal Procedures (RO) Electrical

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
5.3GRID System Lesson COR001-13-01 Main Generator and Auxiliaries

Related Skills (K/A)	
700000.AA1.01	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) Grid frequency and voltage (3.6 / 3.7)

QUESTION: 48

A Main System Grid disturbance is causing voltage and frequency changes on Cooper's Main Generator and 5.3GRID is entered. The current plant conditions are noted:

- Reactor power 95%.
- Main Generator frequency is 59.8 MHz.
- Main Generator voltage is 21.60 KVolts.
- Main Generator MVARs are – 160 MVARs.

What actions should be taken on Panel C to correct the electrical transient?

a. Raise voltage

b. Lower voltage

c. Raise frequency

d. Lower frequency

ANSWER: 48

a. Raise voltage

Explanation:

From the Main Generator Lesson:
Reactive Load - Once the generator output breakers are closed, the generator's reactive load is controlled by the Main Generator field current. When the Gen Voltage Adjust control switch is in the OFF or TEST position, field current is adjusted with the Base Adjuster control switch. With the Gen Voltage Adjust switch in the ON position, field current is adjusted by the voltage adjust control switch.

The lowermost line shows the limit of reactive load (MVARs) when the generator is being operated "underexcited", that is to say that the generator has become a reactive load to the grid, due to the additional current draw from the grid needed to supplement the rotor field in this condition where the regulator is not supplying enough field current on its own. (This is also known as the "vars in" or "leading power factor" region of the curve.) In this condition, part of the rotor field strength is induced by current in the stator. The lines of flux on the stator in this condition are highly concentrated on the stator core ends. **High current flows result in the stator (core) ends, which cause localized overheating to occur. This heating in the stator core ends (retainer sections) is the basis for the limits of the curve in this region.**

From 5.3GRID
Grid oscillations can be induced by CNS generator regulator malfunction or other grid generators malfunctioning. Grid generated oscillations could affect 345 kV, 161 kV, and/or 69 kV voltages. When determining if oscillating voltage has exceeded the lower limit, average oscillating voltage is used.

NPPD System frequency < 59.5 indicates NPPD system is unstable and outside of its secure operating state. If frequency is degraded, it will likely be a rapidly occurring event that will cause automatic system load shedding to stabilize system.

Maintain generator MVAR per DCC System Operator direction and Main Generator MEGAWATTS vs. MEGAVARS per Attachment 2 (Page 8) with GEN BASE ADJUST by performing following: To pick up positive (OUT) MAIN GENERATOR MVAR. Maintain MEGAWATTS vs. MEGAVARS in area bound by: 0.85 LAGGING PF line (positive or over-excited). 0.95 LEADING PF line (negative or under-excited).

Distractors:

b. Lower voltage – would make the situation worse by decreasing MVARs even further and cause more instability in the generator.

c. Raise frequency – with the Frequency at 59.8 MHz, this is still within the normal operating range for frequency and no adjustments need to be made.

d. Lower frequency – with the Frequency at 59.8 MHz, this is still within the normal operating range for frequency and no adjustments need to be made.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295016.AA1.02	
Importance Rating	2.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.1ASD 13

Learning Objective: See Attached (As available)

Question Source:
Bank # 19280
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	19280	01	12/17/2004	05/23/2010	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320134, OPS CNS Abnormal Procedures - Fire

Related Lessons	
INT0320134	OPS CNS Abnormal Procedures (RO) Fire

Related Objectives	
INT0320134H0H0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References
NONE

Related Skills (K/A)	
295016.AA1.02	Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.7 / 45.6) Reactor/turbine pressure regulating system (2.9* / 3.1*)

QUESTION: 49 19280 (1 point(s))

The plant is at power when the following events occur:

- A fire occurs in the Control Room.
- Control Room is evacuated.
- Alternate Shutdown Room is manned.
- A 90°F/hr cooldown is commenced from normal operating pressure.

What indication must the ASD operator use to initially monitor RPV cooldown rate during a shutdown outside the Control Room?

a. RPV pressure on Rack 25-5 and 25-6.

b. Reactor Recirc Suction Temperature on Rack 25-5 and 25-6.

c. HPCI Turbine Steam pressure on the ASD HPCI Control Panel.

d. HPCI Pump Discharge pressure on the ASD HPCI Control Panel.

ANSWER: 49 19280

c. HPCI Turbine Steam pressure on the ASD HPCI Control Panel.

Explanation:

HPCI Turbine Steam Pressure is used to calculate cooldown during these conditions.
Reference: 5.4FIRE-SD Attachment 1 Caution prior to step 2.1.3.

Distractors:

a. is incorrect. Reactor pressure indication on Rack 25-5 is utilized to calculate cooldown rate after HPCI is isolated and RHR is in SDC. These indications are used in 5.3SBO also.

b. is incorrect. Reactor pressure on Rack 25-5 is used after HPCI is isolated on low pressure, not recirc suction temperatures.

d. is incorrect. HPCI pump discharge pressure will be dependent upon HPCI pump speed and not representative of reactor pressure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030.EA1.05	
	Importance Rating	3.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number)

EOP 3A	EOP SAG Graphs	
13	14	

Learning Objective: See Attached (As available)

Question Source:

Bank #	9724	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	H

10 CFR 55 Content

55.41	7
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	9724	00	04/24/2000	05/23/2010	None	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
None	COR0021102001100N

Related Lessons
COR0021102 OPS High Pressure Coolant Injection System

Related Objectives
COR0021102001100N Predict the consequences of the following on the HPCI system: Low suppression pool level

Related References
EOP SAG Graphs

Related Skills (K/A)	
295030.EA1.05	Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.7 / 45.6) HPCI (3.5 / 3.5)

QUESTION: 50 9724 (1 point(s))

The plant is recovering from a scram and the following conditions exist:

- Control rods fully inserted
- RPV water level is -180 inches on Fuel Zone Instrument on Panel 9-3 (slowly lowering)
- RPV pressure is 800 psig
- CRD is injecting to RPV at maximum flow
- HPCI is injecting to RPV at rated flow
- No other sources of RPV injection available
- Primary containment water level is 10.9 feet (lowering)

What action is required?

a. Remove HPCI from service.

b. Emergency depressurize the RPV.

c. Raise injection from HPCI, disregarding Vortex Limits.

d. Enter the Steam Cooling Procedure when RPV water level drops to -125 inches Corrected FZ.

ANSWER: 50 9724

a. is correct. Step SP/L-10 of the Primary Containment Control Procedure because the HPCI turbine exhaust is not submerged when torus level drops below 11 feet.

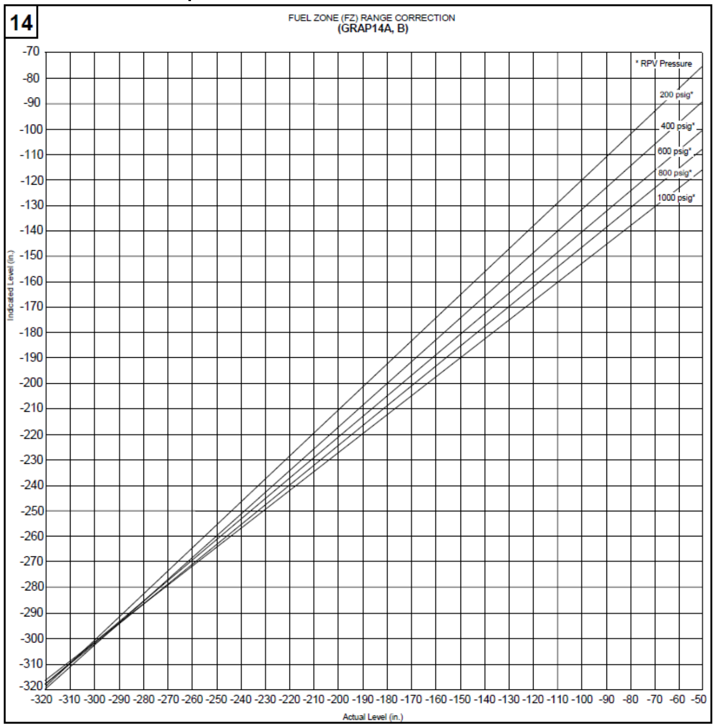
b. is incorrect. Because ED is not required until the RPV water level drops below -158 inches corrected FZ.

c. is incorrect. Because the HPCI turbine is exhausting directly into the torus air space and there is no Vortex Limit for the HPCI pumps.

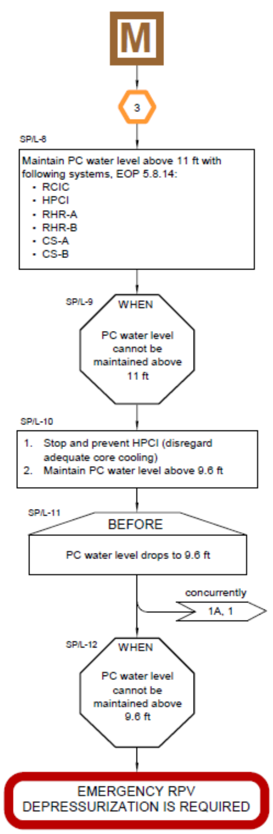
d. is incorrect. Because the operating crew will go to Steam Cooling ONLY when there is no vessel injection available.

REFERENCE: EOP Flowchart 3A, Step SP/L-10, EOP Graph 1

EOP/SAG Graphs



EOP-3A



Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026.EA2.02	
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) EOP/SAG Graphs
(Attach if not previously provided)
(including version/revision number) 14

Learning Objective: See Attached (As available)

Question Source:			
	Bank #	24497	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	10
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	24497	02	03/22/2010		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613 obj 11 & INT0080618 obj 3, EOP 3A, pressure for HCTL (2008 BIENNIAL EXAM)

Related Lessons	
INT0080613	OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL
INT0080618	OPS EOP AND SAG GRAPHS AND CAUTIONS

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.

Related References	
INT0080613	Flowchart 3A - Primary Containment Control
INT0080618	EOP and SAG Graphs and Cautions
5.8	Emergency Operating Procedures (EOPs)

Related Skills (K/A)	
295026.EA2.02	Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool level (3.8 / 3.9)

QUESTION: 51 24497 (1 point(s))

The plant has just experienced a small break LOCA resulting in the following conditions:

- Reactor Pressure 800 psig (steady)
- Torus pressure is 5 psig (rising slowly)
- Primary Containment water level is 11 feet (steady)
- Torus water temperature is 180° F (rising)

In accordance with EOP 3A; what is the **HIGHEST** Torus Water Temperature before the plant is required to be Emergency Depressurized?

a. 185°F

b. 195°F

c. 210°F

d. 263°F

ANSWER: 51 24497

b. 195°F

Explanation:

In accordance with EOP 3A Torus Temperature leg, Step SP/T-5 When average torus water temperature and RPV pressure cannot be maintained within HCTL (Graph 7) EMERGENCY DEPRESSURIZE. At 800 psig and a primary containment level of 11 feet the lines of the HCTL graph intersect at 200°F.

Distractors:

a. 185°F is the temperature for 1080 psig the next line before the correct one.

c. 210°F is the temperature for 600 psig the next line after the correct one.

d. 263°F is the temperature for 50 psig the last line before it turns red.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000.AA2.05	
	Importance Rating	2.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Procedure 2.2.84
(Attach if not previously provided)
(including version/revision number) 49

Learning Objective: See Attached (As available)

Question Source: Bank # 3724
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	3724	04	12/12/2008		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	SKL0124108A0A030E HEATING, VENTILATION, AIR CONDITIONING

Related Lessons
SKL0124108 HEATING, VENTILATION, AIR CONDITIONING COR0010802 OPS HEATING, VENTILATION AND AIR CONDITIONING

Related Objectives	
SKL012410800A030E	Given plant conditions, predict changes in the following: Starting/stopping of fans
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

Related Skills (K/A)	
600000.AA2.05	Ability to determine and interpret Ventilation alignment necessary to secure affected area as they apply to PLANT FIRE ON SITE (2.9 / 3.0)

QUESTION: 52 3724 (1 point(s))

A fire has occurred in the Cable Spreading Room with the plant initially operating at 100% power. The fire did not spread beyond the Cable Spreading Room.

What impact does this fire have on the Control Room Ventilation System?

a. Fire Dampers isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

b. Fire Dampers **DO NOT** isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

c. Fire Dampers isolate the Control Room. The Supply Fans **DO NOT** trip. The Emergency Bypass Train starts and supplies the Control Room with outside air.

d. Fire Dampers **DO NOT** isolate the Control Room. The Supply Fans **DO NOT** trip. The Emergency Bypass Train starts and supplies the Control Room with outside air.

ANSWER: 52 3724

a. Fire Dampers isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

Explanation:

Procedure 2.2.84 ATTACHMENT 1 INFORMATION SHEET
FUNCTION
The system provides HVAC to the Control Room and Cable Spreading Room for personnel comfort and optimum equipment performance .

OPERATING CHARACTERISTICS
1 .2 .4 Fire/Smoke Dampers HV-AD-AD1544, HV-AD-AD1545, HV-AD-AD1546, HV-AD-AD 1547, HV-AD-AD 1581, and HV-AD-AD 1582 automatically close when fire or smoke is detected locally at the damper or when smoke is detected in the Cable Spreading Room to prevent smoke from spreading to the Control Room when there is a fire in the Cable Spreading Room.

When smoke is detected by SD-1001 (Cable Spreading Room return duct), Supply Fans SF-C-1A and SF-C-1B receive trip signals and fire/smoke Dampers HV-AD-AD1544, HV-AD-AD1545, HV-AD-AD1546, HV-AD-AD1547, HV-AD-AD1581, and HV-AD-AD1582 close.

Distractors:

b. The fire dampers do isolate the control room.

c. The Supply Fans trip.

d. The fire dampers do isolate the control room.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001.AA2.05	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-21-02, 20

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis H

10 CFR 55 Content
- 55.41 10
- 55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
53	00	09/29/10			RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	What action is required for a Jet Pump Failure?

Related Lessons
COR0022102 OPS Reactor Protection System

Related Objectives	
COR0022202001060G	Given a specific Reactor Recirculation system or the Recirculation Flow Control system malfunction, determine the effect on any of the following: Reactor Vessel Internals (jet pumps, stratification, bottom head drain temperature, pump starts)

Related References		
COR0022102	OPS Reactor Protection System	Rev20

Related Skills (K/A)		ROI	SROI
295001.AA2.05	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.10 / 43.5 / 45.13) Jet pump operability: Not-BWR-1&2 (3.1/3.4)	3.1	3.4

QUESTION: 53

The reactor is at 100% power when Reactor Power lowers unexpectedly. The following conditions are noted:

- Generator output has decreased.
- Indicated Total Core flow has increased.
- Core plate D/P has decreased
- RR pump speeds have not changed
- RR Loop “A” indicated flow $34x10^6$ lbm/hr and stable
- RR Loop “B” indicated flow $36x10^6$ lbm/hr and stable

What action is required?

a. Lock the “A” Reactor Recirculation Pump Scoop Tube.

b. Perform Jet Pump operability

c. Lower the speed of the “B” RR Pump to within 5% of the “A” RR Pump

d. Raise the speed of “A” RR Pump to within 5% of the speed of the “B” RR Pump.

ANSWER: 53

b. Perform Jet Pump operability

Explanation:

Core flow has risen with no change of recirc pump speeds this does NOT comply with the reactor power and core plate *d/p* lowering; which indicates a lowering of core flow. Recirc pump speed has not changed therefore it appears recirc pump B flow has risen while its speed remained the same this indicates a failed jet pump. For these indications 2.4RxPWR Attachment 2 Step 1.3 requires performing jet pump operability.

Distractors:

a. is incorrect because there has been no change in recirc pump speeds. In accordance with 2.4RR, the scoop tube is locked when recirc pump speed is rising.

c. is incorrect because the requirement is to balance speeds is within 10% when core flow is greater than or equal to 70% rated core flow. There is not a 10% imbalance.

d. is incorrect because the speed should be lowered on the faster pump additionally the requirement is to balance speeds is within 10% when core flow is greater than or equal to 70% rated core flow. There is not a 10% imbalance.

Examination Outline Cross-Reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295006	G	2.4.45
	Importance Rating	4.1		

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 2.4TURB
(Attach if not previously provided)
(including version/revision number) 24

Learning Objective: See Attached (As available)

Question Source:
Bank #
Modified Bank # 12134 (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54		00	12/12/2010		LOR	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	

Topic Area	Description
None	INT0320127O0O0100 CNS Abnormal Procedures (RO) Turbine/Generator

Related Lessons	
INT0320127	CNS Abnormal Procedures (RO) Turbine/Generator

Related Objectives	
INT0320127O0O0100	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	
2.4TURB	Main Turbine Abnormal
10CFR55.41	Written examinations: Operators

Related Skills (K/A)
295006. SCRAM Generic 2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12) (4.1 / 4.3)

QUESTION: 54 (1 point(s))

Power ascension is in progress with reactor power at 25%.

- 9-5-2/C-4, TSV & TCV CLOSURE TRIP BYP is clear
- The Main Turbine is being placed in service and is currently rotating at 1800 RPM.
- Rising thrust bearing metal temperatures are noted.
- Thrust bearing metal temperature on computer point T079 is 220°F and T080 is 230°F.
- The crew also noted that lube oil cooler outlet temperature is 145°F.

Which action is required next?

a. Trip the Main Turbine only.

b. Commence a normal shutdown.

c. Scram the reactor and trip the Main Turbine.

d. Send a Station Operator to reduce lube oil cooler outlet temperature.

ANSWER: 54

a. Trip the Main Turbine only.

Explanation:

Since power is below the set point for 9-5-2/C-4, TSV & TCV CLOSURE TRIP BYP annunciator a reactor scram is not required however the Main Turbine must be tripped in accordance with 2.4TURB.

Distractors:

b. A normal shutdown will be commenced if the temperature is not corrected; however the turbine trip must be performed as the set points have been exceeded.
c. A turbine trip is required; however the Reactor need not be scrammed.
d. A turbine trip is required; sending a station operator to investigate the problem is a little late.

Modified: 12134

QUESTION: 12134

At 2230 a power ascension was in progress with reactor power at 90%. Rising thrust bearing metal temperatures were noted. Thrust bearing metal temperature on computer point T079 was 220°F and T080 is 230°F. The crew also noted that lube oil cooler outlet temperature was 145°F.

Which action is required next?

a. Reduce turbine load by 10%.

b. Commence a normal shutdown.

c. Scram the reactor and trip the turbine.

d. Reduce lube oil cooler outlet temperature.

ANSWER:

c. is correct. Since power remains above the setpoint for 9-5-2/C-4, TSV & TCV CLOSURE TRIP BYP annunciator a reactor scram and then a turbine trip are required.

a. is incorrect. Because a turbine trip is required.
b. is incorrect. Because a turbine trip is required.
d. is incorrect. Because a turbine trip is required.

Examination Outline Cross-Reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295038	G	2.2.44
	Importance Rating	4.2		

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Procedure 5.7.17
(Attach if not previously provided)
(including version/revision number) 35

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	13
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55		00	01/02/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	3	Multiple Choice	

Topic Area	Description
Emergency Procedures	

Related Lessons
GEN0030401 Emergency Plan for Licensed Operators

Related Objectives
GEN0030401E0E0200 State the primary method used to quantify the source term. GEN0030401E0E0300 List the monitored release path at CNS.

Related References
5.7.17 Dose Projection

Related Skills (K/A)
295038 High Off-site Release Rate Generic 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) IMPORTANCE RO 4.2 SRO 4.4

QUESTION: 55

The reactor has scrammed due to a steam leak into Secondary Containment, with the following conditions:

- RMP-RM-452A B, C, D, RX BLDG VENT RAD MONITORS are all reading 30 mRem.
- MAX Safe Temperatures and Radiation levels are exceeded in one (1) area.

The Operator designated to perform Emergency Procedure 5.7.17 Dose Projection gets to Step 5.2 which asks to determine if the SGTs are in the effluent stream.

How should the Operator answer the question and why?

a. The SGTs are not in the effluent path, because they automatically trip under these conditions.

b. The SGTs are not in the effluent path, Operator action is required to place them in the path during these conditions.

c. The SGTs are in the effluent path, because a Group 2 has caused an isolation of the Reactor Building and the start of both trains.

d. The SGTs are in the effluent path, because a Group 6 has caused an isolation of the Reactor Building and the start of both trains.

ANSWER: 55

d. The SGTs are in the effluent path, because a Group 6 has caused an isolation of the Reactor Building and the start of both trains.

Explanation:

With a Primary System discharging into the Reactor Building and all four Reactor Building Exhaust Ventilation Rad Monitors reading above their set points for a Group 6 isolation, Both SGTs should be running taking a suction on the Reactor Building.

Distractors:

a. Both SGTs get a start signal from the Rx Building Vent Rad Monitors Hi-Hi of 11 mR/hr
b. Both SGTs get a start signal from the Rx Building Vent Rad Monitors Hi-Hi of 11 mR/hr
c. Both SGTs get a start signal from the Rx Building Vent Rad Monitors Hi-Hi of 11 mR/hr, which is a Group 6 signal not a Group 2.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004	G 2.1.26
	Importance Rating	3.4	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.3DC125 0.36.8 Electrical Safety Handbook

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56		00	01/12/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320131, Loss of DC reason the reactor scrams.

Related Lessons	
INT0320131	CNS Abnormal Procedures (RO) Electrical

Related Objectives	
INT0320131T0T0100	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).

Related References	
5.3DC125	Loss of 125 VDC
0.36.8	Electrical Safety Handbook

Related Skills (K/A)
2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (CFR: 41.10 / 45.12) IMPORTANCE RO 3.4 SRO 3.6

QUESTION: 56

A loss of 125 VDC Bus A occurs. It is determined that 125 VDC Bus A is not being supplied by the battery bank due to loose connections. One electrician starts to torque the positive bus terminals and the other electrician starts to torque the negative terminals on the Electrical bus. Both electricians are within reach of each other and are sharing tools by handing them back an forth. And both are wearing the appropriate PPE.

Does this meet CNS Electrical Safety standards and if not what should be done to comply?

a. Yes.

b. No; Work cannot be performed on any energized electrical equipment.

c. No; the battery terminals must be disconnected first so their work is performed on de-energized equipment.

d. No; the electricians need to be separated so that they cannot form a circuit by touching each other and opposite poles in the bus.

ANSWER: 56

d. No, the electricians need to be separated so that they cannot form a circuit by touching each other and opposite poles in the bus.

Explanation:

ADMINISTRATIVE PROCEDURE 0 .36.8 ELECTRICAL SAFETY RULE BOOK
ATTACHMENT 4 GENERAL REQUIREMENTS
All circuits and equipment operating > 50V to ground should be de-energized prior to beginning work, unless:
De-energizing is infeasible or impractical due to equipment design, operational limitations, or requirements of the work activity

When two or more individuals are working on exposed energized equipment and are within reach of each other, they shall not work on different phases/polarity at the same time (e.g., battery systems).

Distractors:

a. In accordance with 0.36.8 Electrical Safety Rule Book, working within reach of each other is not allowed when working on different phases/polarity such as batteries.

b. Some systems cannot be de-energized such as the batteries, therefore provisions in the safety manual are made, and as long as the precautions are met, work may be done.

c. Energized work is energized work, no matter where it is. There is no way to de-energize a battery.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025.EA2.02	
	Importance Rating	4.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 2.3_9-5-2

Learning Objective: See Attached (As available)

Question Source:		
Bank #	19077	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam: NA

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

10 CFR 55 Content	
55.41	10
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	19077	02	12/31/2008		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0022102, RPS Main Turbine trip with MTBVs initially open

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001040J	Describe the RPS design features and/or interlocks that provide for the following: Bypassing of selected scram signal (manually and automatically)
SKL012422100A030I	Given plant conditions, predict changes in RPS components/parameters: Reactor power.

Related References	
2.3_9-5-2	Panel 9-5 - Annuciator 9-5-2

Related Skills (K/A)	
295025.EA2.02	Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Reactor power (4.2* / 4.2)

QUESTION: 57 19077 (1 point(s))

With the plant initially at 35% power and all conditions normal, a voltage surge in the DEH control system results in the following plant conditions:

- Main Generator load is 192 MWe.
- Main Turbine Bypass Valves are partially open.
- Total Main Steam flow is 35%.
- Annunciator 9-5-2/C-4, TSV & TCV CLOSURE TRIP BYP CHAN A/B, is in alarm.
  1. (2704) TSV & TCV CLOSURE TRIP BYPASSED CHAN A1
  2. (2705) TSV & TCV CLOSURE TRIP BYPASSED CHAN A2
  3. (2706) TSV & TCV CLOSURE TRIP BYPASSED CHAN B1
  4. (2707) TSV & TCV CLOSURE TRIP BYPASSED CHAN B2

What is the expected plant response if a Main Turbine trip should subsequently occur?

a. Reactor continues to operate at 35% power.

b. Reactor continues to operate at 25% power.

c. Reactor scrams on high reactor pressure.

d. Reactor scrams on TSV closure.

ANSWER: 56 19077

c. Reactor scrams on high reactor pressure.

Explanation:

Since the TSV/TCV closures are bypassed due to the bypass valves being open a turbine trip will not initiate a scram but will increase reactor pressure causing on a scram on high reactor pressure.

Distractors:

a. is incorrect. When the turbine trips reactor power is greater than bypass valve capacity, so reactor power cannot continue steady at 35% power.

b. is incorrect. Reactor power will not automatically lower to bypass valve capacity; operator action must be taken to reduce reactor power.

d. is incorrect. The TSV closure scram signal is bypassed as indicated by annunciator 9-5-2, C-4.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031.EK1.01	
	Importance Rating	4.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Flowchart 2A Emergency RPV Depressurization

Learning Objective: See Attached (As available)

Question Source:
Bank # 12325
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR 55 Content
55.41 8
55.43 ______

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	12325	02	03/28/2006		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	H	1	6	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080607, FLOWCHART 2A - EMERGENCY RPV DEPRESSURIZATION/ STEAM COOLING

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.
INT00806070010700	Given plant conditions and EOP flowchart 2A, EMERGENCY RPV DEPRESSURIZATION/STEAM COOLING, determine required actions.

Related References	
10CFR55.41	Written examinations: Operators
INT0080607	Flowchart 2A Emergency RPV Depressurization

Related Skills (K/A)	
295031.EK3.04	Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.5 / 45.6) Steam cooling.(4.0/4.3*)
295031.EK1.01	Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.8 to 41.10) Adequate core cooling. (4.6*/4.7*)

QUESTION: 58 12325

The plant was operating at power when a station blackout occurred. HPCI and RCIC will not inject into the RPV. The crew is able to operate RCIC in a pressure control mode. Reactor water level lowered to -158 inches (corrected FZ) and the crew entered steam cooling. RPV pressure stabilized using the RCIC system at a reactor pressure of 800 psig.

Five minutes later, the following plant conditions were present:

- Reactor water level -183 inches (corrected FZ) and slowly lowering
- Reactor pressure 810 psig
- Average drywell temp 210ºF
- RCIC turbine speed 4500 RPM
- RCIC Flow 500 gpm

What action is required and why?

a. Emergency depressurize because the core may not be adequately cooled.

b. Alternate emergency depressurize to preserve post depressurization inventory.

c. Continue to operate RCIC because adequate core cooling is assured until reactor level reaches -202 inches (corrected FZ).

d. Allow pressure to rise and be controlled by SRV pressure relief setpoint(s) to ensure adequate steam flow to cool the core.

ANSWER: 58

c. Continue to operate RCIC because adequate core cooling is assured until reactor level reaches -202 inches (corrected FZ).

Explanation:

Emergency Depressurization is required at -202 inches FZ (corrected). Operation of RCIC is allowed to slow the pressure rise allowing continued steam cooling. Boil-off is occurring while level is slowly lowering maximizing the time available for steam cooling to occur.

Distractors:

a. -183 inches is the level you emergency depressurize at if you have an injection source available. In this case you do not so you are directed to wait until level drops to -202 inches.

b. Alternate emergency depressurize is not allowed if you should wait for -202 inches to perform an emergency depressurization.

d. If RCIC is still incapable of injection, there is no reason to remove it from pressure control and allow pressure to rise to the SRV setpoints.

Provide to Candidate: **EOP flowchart 2A with cautions removed.**

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295008.AK3.03	
	Importance Rating	3.4	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Tech Specs Bases 3.3.2.2

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	5
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59		00	01/12/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency	What is the reason for the high level trip of the main turbine?

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL INT0320135 CNS Abnormal Procedures (RO) - Condensate/Feedwater

Related Objectives	
COR0023202001090D	Given a specific RVLC system malfunction determine the effect on any of the following: RPV water level
INT0320135H0H0100	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).

Related References
Tech Specs Bases 3.3.2.2

Related Skills (K/A)	
295008.AK1.03	Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.5 / 45.6) Main turbine trip (3.4 / 3.5)

QUESTION: 59

Why does the Main Turbine trip on a high reactor water level?

a. Provide anticipatory Scram to prevent exceeding MCPR safety limit.

b. Provide anticipatory Main Turbine trip when both Reactor Feed Pumps trip.

c. To prevent moisture carryover into the Main Turbine that could cause turbine blade damage.

d. To reduce the load on the Reactor to allow the feedwater system to attempt to recover level before scramming.

ANSWER: 59

c. To prevent moisture carryover into the Main Turbine that could cause turbine blade damage.

Explanation:

Tech Specs 3.3.2.2 Bases - BACKGROUND - The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level, Level 8 reference point, causing the trip of the two feedwater pump turbines and the main turbine.
Reactor Vessel Water Level-High, Level B signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level-High, Level B instrumentation are provided as input to a-two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. Each channel consists of a level transmitter loop and a trip relay that compares measured input signals with pre-established-setpoints. When the setpoint is exceeded, the channel outputs a main feedwater and main turbine trip signal to the trip logic. A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop and control valves protects the turbine from damage due to water entering the turbine.

Distractors:

a. Although tripping the main turbine provides an anticipatory Scram on high water level it is not there to prevent exceeding MCPR safety limit. The Reactor Feed Pump trip to provide this function.

b. There is no anticipatory Main Turbine trip on a trip of both Reactor Feed Pumps. This answer is a combination of two of the other answers that are incorrect.

d. There is a Reactor Scram on the Turbine trip so there is no way the feedwater level control system can recover prior to scramming.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002.AK2.08	
	Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR001-02-01

Learning Objective: See Attached (As available)

Question Source:
Bank # 2590
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	2590	00	01/03/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0010202001040A Circulating Water

Related Lessons
COR0010202 OPS Circulating Water

Related Objectives
COR0010202001040A Given a specific Circulating Water malfunction, determine the effect on any of the following: Main Condensate System

Related References	
COR0010202	OPS Circulating Water

Related Skills (K/A)
295002.AK2.08 Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) Condenser circulating water system 3.1 / 3.2

QUESTION: 60 2590 (1 point(s))

The plant is operating at 60% power when one of the operating circulating water pumps trip.

What is the effect on the plant when this pump trips?

a. Main Generator output rises.

b. Condenser vacuum will degrade.

c. Overall plant efficiency improves.

d. Condensate Pump NPSH increase.

ANSWER: 60 2590

b. Condenser vacuum will degrade.

Explanation:

At 60% power the loss of one Circulating Water Pump will cause cooling to decrease in the Main Condenser effectively lowering condenser vacuum.

Distractors:

a. The turbine efficiency decreases therefore causing an output reduction

c. The turbine efficiency decreases therefore overall plant efficiency lowers

d. The turbine efficiency decreases therefore causing an increase in the hotwell temperatures and a reduction of NPSH to the Condensate Pumps

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295015.AK3.01	
	Importance Rating	3.4	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson COR002-26-02 , 5.8.3
18 9

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 5
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	New	00	01/07/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	Rod insertion during an ATWS why bypass the insertion blocks?

Related Lessons
INT0080606 FLOWCHART 6A - RPV PRESSURE/POWER (FAILURE-TO-SCRAM)

Related Objectives
INT00806060010600 List the methods of alternate rod insertion.

Related References	
PRO	5.8.3
LP	COR002-26-02

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: 61 (1 point(s))

The Plant was operating at near rated power when a steam line rupture in the HPCI room occurred. A manual scram was attempted. An ATWS occurred, with the following conditions:

- 25 control rods are at position 24 or GREATER.
- All scram valves are open.

The CRS has directed you to enter 5.8.3, ALTERNATE ROD INSERTION METHODS and insert control rods with RMCS.

Why is the Rod Worth Minimizer bypassed for this evolution?

a. To prevent all Rod Block Monitor rod blocks.

b. To prevent inadvertent rod withdrawals during the ATWS.

c. To allow the Control Rods to be driven in with the Rod Movement Control Switch.

d. To allow the Control Rods to be selected on the select matrix and driven in with the Emergency Notch Override Control Switch.

ANSWER: 61

d. To allow the Control Rods to be selected on the select matrix and driven in with the Emergency Notch Override Control Switch.

Explanation:

In accordance with Emergency Procedure 5.8.3 the RWM is bypassed in step ARI-20 to allow the Rod Insertion Blocks to be cleared so the emergency notch override switch can be used to insert the control rods.

Distractors:

a. Bypassing the Rod Worth Minimizer does not prevent the rod blocks from the RBM.

b. Bypassing the Rod Worth Minimizer will not prevent the operator from driving the control rod in the wrong direction (i.e. OUT)

c. The control rods are not driven with the normal Rod Movement Control Switch in the event there is an ATWS, Emergency Notch Override Switch is positioned in the INSERT direction.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032.EA1.05	
	Importance Rating	3.7	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.1.22
55

Learning Objective: See Attached (As available)

Question Source: Bank # 2622
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	2622	01	08/24/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	3	Multiple Choice	

Topic Area	Description
Systems	COR0020302001060D Containment

Related Lessons
COR0020302 OPS CONTAINMENT

Related Objectives
COR0020302001060D Describe the interrelationship between PCIS and the following: HPCI

Related References	
PR	2.1.22

Related Skills (K/A)	
295032.EA1.05	Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.7 / 45.6) Affected systems so as to isolate damaged portions (3.7 /3.9)

QUESTION: 62 2622 (1 point(s))

The HPCI System auto initiated but shortly after the following conditions were indicated.

- HPCI System Flow .... Normal
- Reactor Level ....+20 inches
- HPCI Area Temperature ....205°F
- D/W Pressure .... 1.2 psig
- HPCI Steam Supply Pressure .. 925 psig

What automatic actions will occur and why?

a. A Group 4 isolation from steam line low pressure.

b. A Group 4 isolation from high area temperature.

c. A Group 5 isolation from high area temperature.

d. A Group 5 isolation from high drywell pressure.

ANSWER: 62 2622

b. A Group 4 isolation from high area temperature.

Explanation:

With the HPCI Area Temperature 205°F this has exceeded the trip setpoint for the isolation. Group 4 is HPCI valves and Group 5 is RCIC.

Distractors:

a. HPCI Steam Supply Low pressure isolation is approximately 100 psig we are well above that.

c. Group 5 are the RCIC Valves not the HPCI ones, RCIC does not isolate on a HPCI Steam Line high area temperature.

d. Group 5 are the RCIC Valves not the HPCI ones, RCIC does not isolate on high drywell pressure.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022.AA2.01	
	Importance Rating	3.5	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Tech Specs 3.1.5
(Attach if not previously provided)
(including version/revision number) Amendment 178

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam Pilgrim 2009

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

10 CFR 55 Content		
	55.41	10
	55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63		00	01/07/11		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070502, CNS Tech Spec 3.1, Reactivity Control Systems

Related Lessons
INT0070502 CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT00705020010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.1 LCO, determine the ACTIONS that are required.

Related References	
3.1.5	Control rod scram accumulators

Related Skills (K/A)	
295022.AA2.01	Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: (CFR: 41.10 / 43.5 / 45.13) Accumulator pressure (3.5 / 3.6)

QUESTION: 63

A Plant startup is in progress with the following conditions:

- The Reactor MODE Switch is in the Startup position.
- Procedure 2.1.1 is at the point of placing the first Reactor Feed Pump in service.
- CRD Pump “A” is out of service for bearing replacement.

When the following occurs:

- CRD Pump “B” trips on overcurrent
- CRD ACCUM LOW PRESS OR HIGH LEVEL Alarms
- Full Core display indicates that there are two (2) accumulator alarms on control rods that are at position 48.
- The Station Operator verifies that they are caused by low pressure.

In accordance with Technical Specifications 3.1.5 “Control Rod Scram Accumulators” what action is required?

a. Declare the associated control rods inoperable within 1 hour.

b. Declare the associated control rod scram times slow within 1 hour.

c. Place the Reactor MODE Switch in the shutdown position immediately.

d. Restore CRD Charging Header pressure to > 940 psig within 20 minutes.

ANSWER: 63

c. Place the Reactor MODE Switch in the shutdown position immediately.

Explanation:

With the plant at the point of placing the first feed pump in service, Reactor Pressure should be around 600 psig in accordance with the Startup Procedure. When the running CRD Pump trips and charging header pressure drops below 940 psig, Tech Specs 3.1.5 Condition C states to verify that the associated control rods are fully inserted. In this case they were full out. So condition C could not be met and the operator is required to enter condition D, that states to place the reactor mode switch in the shutdown position immediately.

Distractors:

a. Only allowed by Tech Specs if Reactor Steam pressure is above 940 psig.
b. Only allowed by Tech Specs if Reactor Steam pressure is above 940 psig.

d. Only allowed by Tech Specs if Reactor Steam pressure is above 940 psig.

NRC EXAM - 2009

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	295022	AA2.01
	Importance Rating	3.5	

K&A: Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: Accumulator Pressure

Loss of CRD Pumps

Explanation: **Answer B** per TS 3.1.5 required action for reactor pressure < 600 psig. (RFBP to MFP shift @ 250 psig max)

A – Per TS 3.1.5, 1520 psig – ONI-C11-1 uses 1600psig for readability

C – incorrect – correct if > 600 psig

D – incorrect – action for declaring rod slow

Technical Reference(s): TS 3.1.5, ONI-C11-1 Rev 10	Reference Attached: TS 3.1.5 pp 3.1-15 to 17, ONI-C11-1 p 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-3037-05.D

Question Source:	Bank # Modified Bank # New	 Perry static 09-455
Question History:	Previous NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X
10 CFR Part 55 Content:	55.41 55.43	X

Comments: Level of Difficulty = 3

QUESTION RO 37

Plant startup is in progress. The following conditions exist:

- Mode Switch in STARTUP
- Feed water shift in progress from RFBPs on Low flow controller to the Motor Feed Pump
- CRD 'A' pump is out of service for bearing replacement
- CRD 'B' pump trips on over current
- 2 accumulator faults come in on control rods at position 48

Per TS 3.1.5 Control Rod Scram Accumulators, the following is correct with respect to current plant conditions?

A. 2 accumulators @ 1600 psig would require the Mode Switch to be placed in SHUTDOWN immediately

B. 1 accumulator @ 1500 psig and 1 accumulator @ 1600 psig would require the Mode Switch to be placed in SHUTDOWN immediately

C. 2 accumulators @ 1600 psig would require the Mode Switch to be placed in SHUTDOWN within 20 minutes

D. 1 accumulator @ 1500 psig and 1 accumulator @ 1600 psig would require the Mode Switch to be placed in SHUTDOWN within 20 minutes

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295033 G 2.4.9	
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.8.4, Lesson COR002-23-02
16, 27

Learning Objective: See Attached (As available)

Question Source:
- Bank #
- Modified Bank # 1140 (Note changes or attach parent)
- New

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis H

10 CFR 55 Content
- 55.41 10
- 55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	Modified	00	01/08/2011		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Systems	COR00223020011600, COR0022302 Residual Heat Removal

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR00223020011600	Given plant conditions including a Shutdown Cooling isolation, determine actions required to place RHR in the LPCI mode.

Related References	
COR002-23-02	
PR	5.8.4 ALTERNATE INJECTION SUBSYSTEMS

Related Skills (K/A)	
295033 High Secondary Containment Area Radiation Levels	
Generic 2.4.9	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.8 SRO 4.2

QUESTION: 64

The Plant is operating at 40% power when a fuel failure causes the plant to be scrammed and placed in Shutdown Cooling using RHR “B” Pump. The following conditions exist:

- RHR “B” Pump is the only LPCI Pump available.
- RHR-MO-13B Motor breaker tripped and the valve was manually closed when being placed in Shutdown Cooling.
- RHR PUMP ROOM, (SOUTHWEST) RMA-RA-11 is pegged upscale.
- Drywell Rad Monitors RMA-RM-40A & B are reading 200 Rem/hr.
- Technical Support Center (TSC) is not operational yet.

A LOCA occurs on the RPV bottom drain line where water level drops to -145 inches on the Wide Range Instruments.

What Actions are required to recover reactor water level with RHR “B” Loop?

a. Send a Station Operator and a Radiological Technician into the SW quad and manually open the RHR-MO-13B then manually start the “B” RHR pump.

b. Send a Station Operator by himself to the SW quad to manually open the RHR-MO-13B then manually start the “B” RHR pump.

c. Align the control switches for the “B” RHR Pump suction and discharge valves and allow the “B” pump to Auto Start when RPV level reaches -113 inches.

d. Align the control switches for the “B” RHR Pump suction and discharge valves and manually start the “B” pump before RPV level reaches -113 inches.

ANSWER: 64

a. Send a Station Operator and a Radiological Technician into the SW quad and manually open the RHR-MO-13B then manually start the “B” RHR pump.

Explanation:

The “B” RHR Pump will trip when the SDC Suction path is lost, operator action must be taken to realign a suction and injection path to the reactor and then manually start the pump. Since there are high rad conditions in secondary containment near the RHR Pump suction valve, and the ARM is reading off-scale high, All field actions require by the Station Operators have this PRECAUTION in each section of their Emergency Procedures “If dispatched personnel must travel through or work in vicinity of an off-scale Station Area Radiation Monitor, they shall be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician.

Distractors:

b. Since there are ARMs Off-scale high in the area the Station Operator must be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician.

c. The “B” pump will not start with it’s suction valve shut. And the anti-pumping logic in the 4160 breaker will not allow the pump to auto start when RPV level reaches -113 inches.

d. Since there are ARMs Off-scale high in the area the Station Operator must be accompanied by a Radiological Protection Technician or Chemistry/Radiological Protection On-Site Availability Technician. The suction valve must be manually aligned for the “B” RHR Pump.

Modified Question 1140

QUESTION: 1140

Which statement identifies the response of the Residual Heat Removal system (RHR) system if it was operating in SDC and a valid LPCI initiation signal was received?

Consider only the loop operating in SDC.

a. The pump will continue to operate and the valves will realign to inject from the suppression pool into the reactor.

b. The pump will trip, operator action must be taken to realign a suction and injection path to the reactor and then restart the pump.

c. The pump will trip, the valves will realign to inject from the suppression pool into the reactor and then the pump will automatically start.

d. The pump will continue to operate in the SDC mode, operator action is required to stop the pump, realign suction and injection paths and restart the pump.

ANSWER:

b. The pump will trip, operator action must be taken to realign a suction and injection path to the reactor and then restart the pump.

REFERENCE: STCOR002-23-02, page 48, section V.J, rev. 14.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295033	G 2.4.9
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 5.8.4, Lesson COR002-23-02
(Attach if not previously provided)
(including version/revision number) 16 27

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43 ____

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	New	0	01/07/2011		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons
NT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives
NT0080613 OPS EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related References
EOP FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Skills (K/A)
295029 High Suppression Pool Water Level Generic 2.4.1: Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 4.6 SRO 4.8

QUESTION: 65

The plant is operating at 100% power when the suction valve CS-66 for Core Spray Pump A started leaking by.

At what level in the suppression pool will entry into EOP-3A be required?

When the Suppression Pool water level exceeds…

a. + 1.5 inches

b. + 2.0 inches

c. - 1.5 inches

d. - 2.0 inches

ANSWER: 65

b. + 2.0 inches

Explanation:

With the Core Spray Suction Valve leaking, suppression pool level will be rising, because the ECST is at a higher elevation than the water in the torus. As with water level begins to rise +1.5 inches will cause the Torus High Water Level Alarm to sound, however, it is not the entry level for the EOPs. Level has to rise above + 2.0 inches to actually cause entry into EOP 3A Primary Containment Control.

Distractors:

a. This is the high alarm setpoint, not the entry condition.

c. This is the low alarm setpoint, and should not come in because the leaking of the suction valve should cause level in the torus to rise not fall.

d. This is the low entry point for EOP 3A, the leaking of the suction valve should cause level in the torus to rise not fall.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.5	
	Importance Rating	2.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.0.3
72

Learning Objective: See Attached (As available)

Question Source:
Bank # 16465
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43 ____

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	16465	01	01/01/11		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
None	Crew manning

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)
INT0070513	CNS Technical Specifications 5.0, Administrative Controls

Related Objectives	
INT032010300C010H	Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Control Room and Station Shift Staffing Requirements
INT032010300C0400	Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Given a Control Room staffing level, determine if the proper staffing requirements are met.
INT00705130010100	Given a set of plant conditions, recognize non-compliance with a Chapter 5.0 Requirement.
INT00705130010200	Given a set of conditions that constitutes non-compliance with a Chapter 5.0 Requirement, determine the actions that are required.

Related References
Procedure 2.0.3, Conduct of Operations

Related Skills (K/A)	
2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12) IMPORTANCE RO 2.9* SRO 3.9

QUESTION: 66 16465 (1 point(s))

Given the following conditions:

- The plant is in MODE 4
- Shutdown cooling in service
- No work is scheduled

Which one of the following crew compliments meets the MINIMUM requirement for ACTIVE LICENSED OPERATOR personnel required to be physically present in the Control Room?

a. Only 1 licensed operator; must be a SRO.

b. Only 1 licensed operator; either an RO or SRO.

c. 2 licensed operators; must be 2 SROs.

d. 2 licensed operators; must be an RO and SRO.

ANSWER: 66 16465

b. Only 1 licensed operator; either an RO or SRO.

Explanation:

Two active licensed operators (one of which must be an SRO) are required in MODE 4 (Cold Shutdown). Only one licensed operator must be in the Control Room at the controls. This will normally be an RO; however, per section 9, note 2, and higher grade licensed operators may take the place of lower grade licensed operators, therefore, an SRO meets the requirement. REFERENCE: 2.0.3

Distractors:

a. Can be an RO or SRO. An SRO is not required.

c. Only one licensed operator must be in the Control Room at the controls. The operator may believe two licensed operators are required (which is likely because 2 people but not necessarily licensed operators are required in the control room for security purposes). If this was the case, then this could be 2 SROs because step 10.2.4.1, higher grade licensed operators may take the place of lower grade licensed operators, therefore, an SRO meets the requirement.

d. Only one licensed operator must be in the Control Room at the controls. The candidate may believe two licensed operators are required (which is likely because 2 people are required in the control room for security purposes).

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.17	
	Importance Rating	3.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.0.3
72

Learning Objective: See Attached (As available)

Question Source:
Bank # 19155
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	19155	01	12/04/2008		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Administrative	INT0320103, "Round of Containment Parameters" requirements

Related Lessons	
OTH0151003	Focus Area Standards
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
OTH0151003001010D	From memory define the following terms in accordance with in procedure 2.0.3, Conduct of Operations, and Operations Instruction #7: Briefs and updates
INT032010300C010J	Procedure 2.0.3, Conduct of Operations: Discuss the following as described in Conduct of Operations Procedure 2.0.3, Conduct of Operations: Announcing Parameters and Trends

Related References	
2.0.3	Conduct of Operations

Related Skills (K/A)
2.1.17 Ability to make accurate, clear, and concise verbal reports. (CFR: 41.10 / 45.12 / 45.13) IMPORTANCE RO 3.9 SRO 4.0

QUESTION: 67 19155 (1 point(s))

During a LOCA, the CRS requests an update on certain Primary Containment parameters. The RO reports the following to the CRS:

- "Drywell pressure is 4.5 psig and rising slowly."
- "Drywell temperature is 160 degrees and rising."
- "Torus water level is 0 inches and steady."
- "Torus water temperature is 85 degrees and lowering slowly."

Does the report made by the RO meet the requirement(s) for the requested information? Why or why not?

a. Meets. ONLY parameter values are required.

b. Meets. ONLY changing parameters require trends or rates of trends.

c. Does NOT meet. Trends and rates of trends are required for the identified parameters.

d. Does NOT meet. Trends are required for ALL of the identified parameters. Rates of trends are not required.

ANSWER: 67 19155

c. Does NOT meet. Trends and rates of trends are required for the identified parameters.

Explanation:

From Conduct of Operations Procedure 2.0.3, Section 7
All disciplines communicating with the Control Room shall use three (3)-way communication for all orders/directions that involve operation of plant equipment or exchange of critical information related to a given evolution.

When reporting a parameter, the VALUE and TREND shall be given. If the parameter is also outside of the assigned band, the reason should also be stated.

As plant conditions permits and when it can be established, a RATE should be reported with the given parameter.

Distractors:

a. Does not meet standards. Trends and rates of trends are required.

b. Does not meet standards. Trends and rates of trends are required.

d. Does not meet standards. Trends and rates of trends are required.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	2.2.36	
Importance Rating	3.1	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Tech Specs. 3.8.1

Amendment 233

Learning Objective: See Attached (As available)

Question Source:

Bank # 21776

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge L

Comprehension or Analysis ______

10 CFR 55 Content

55.41 10

55.43 ______

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
68 21776	00	08/16/2005		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	6	Multiple Choice	N

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070509, Maintenance on DG Lube oil pump.

Related Lessons	
INT0070509	OPS Tech. Spec. 3.8, Electrical Power Systems

Related Objectives	
INT00705090010100	Given a set of plant conditions, recognize non-compliance with a Section 3.8 LCO.

Related References	
TS	3.8

Related Skills (K/A)		ROI	SROI
2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13) (3.1/4.2) (/)		

QUESTION:

While operating at 100% power, you are informed that Maintenance is planned for the #2 Diesel Motor Driven Lube Oil Pump later this shift.

What LCO is expected due to this activity?

a. 3.8.1 AC Sources Operating **Only**

b. 3.8.2 AC Sources Shutdown **Only**

c. 3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air **Only**

d. 3.8.1 AC Sources Operating **And** 3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air

ANSWER:

a. 3.8.1 AC Sources Operating **Only**

Explanation:

With the plant at 100% power, TS 3.8.1 is the applicable spec to enter, because performing maintenance on the Lube Oil Pump will affect the Diesel's ability to perform its intended function.

Distractors:

b. LCO 3.8.2 is AC Power shutdown so this could not be a correct answer

c. LCO 3.8.3, though it covers Lube Oil, it does only concern itself to purity not the ability to pump the oil.

d. LCO 3.8.3, though it covers Lube Oil, it does only concern itself to purity not the ability to pump the oil.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.2.14	
	Importance Rating	3.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 0.31
59

Learning Objective: See Attached (As available)

Question Source:
Bank # 12203
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 10
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	12203	03	12/04/2008		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	

Topic Area	Description
Administrative	INT0320101, System Component Checklist is being performed; a valve is found out of position. What actions are required for this condition and why?

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

Related Objectives	
INT032010100H010N	Discuss the following as described in Administrative Procedure 0.31, Equipment Status Control: System line-up deviations

Related References	
10CFR55.41	Written examinations: Operators
0.31	Equipment Status Control

Related Skills (K/A)	
2.2.14	Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10 / 43.3 / 45.13) IMPORTANCE RO 3.9 SRO 4.3

QUESTION: 69 12203 (1 point(s))

The plant is in a refueling outage. While the HPCI System Component Checklist is being performed, the independent verification performer finds that HPCI-V-23, MAIN PUMP VENT is open.

What actions are required for this condition?

Immediately notify the Shift Manager and…

a. wait for the SM to direct its closing **only**.

b. write a notification and have the QA group perform an investigation into the out of position valve before closing it.

c. close the valve when directed by the SM, initial the checklist, write a CR, and record the discrepancy on the discrepancy sheet and attach it to the Checklist.

d. close the valve when directed by the SM, initial the checklist, write a CR, notify the QA group supervisor, and record the discrepancy on the discrepancy sheet and attach it to the Checklist.

ANSWER: 69 12203

c. close the valve when directed by the SM, initial the checklist, write a CR, and record the discrepancy on the discrepancy sheet and attach it to the Checklist.

Explanation:

Procedure 0.31 Page 18: SYSTEM LINE-UP DEVIATIONS

When there are multiple positions listed as a Normal position for a component, the AS FOUND position of the component shall be documented. If the AS FOUND position matches the applicable comment, no discrepancy exists. If the AS FOUND position does not match the applicable comment, the actions specified in Step 17.2 shall be taken.

If while performing a Performer Verification or an Independent Verification of a System Component Checklist a component is found in other than the Normal position, perform following:

- Immediately notify the SM.
- Position the component as directed by the SM and initial the System Component Checklist for Performed By.
- The SM shall ensure an Independent/Concurrent Verification is performed.
- Ensure CR is generated addressing all applicable information available.
- Record component description, AS FOUND position and AS LEFT position on a Discrepancy Sheet and ensure it is attached to the System Component Checklist.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.11	
	Importance Rating	3.8	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Lesson INT008-06-17
16

Learning Objective: See Attached (As available)

Question Source:
Bank # 23483
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 12
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
70 23483	01	03/20/2003	05/23/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	N

Topic Area	Description
Emergency Operating Procedures	INT0080617, Why do you restart TB Vent while in 5A? (ILT 2006 NRC EXAM)

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

Related Objectives	
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	
10CFR55.41	(B)(12)

Related Skills (K/A)		ROI	SROI
2.3.11	Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10) (3.8/4.3) (2.7 / 3.2)		

QUESTION: 70

What is the basis for restarting building ventilation in the Turbine Building when executing EOP-5A, RADIOACTIVITY RELEASE CONTROL?

Operation of Turbine Building ventilation…

a. maintains equipment availability AND assures that radioactivity releases pass through a monitored release point.

b. preserves personnel accessibility AND assures that radioactivity releases pass through a monitored release point.

c. maintains equipment availability AND assures a minimum amount of radioactivity plates out on turbine building surfaces.

d. preserves personnel accessibility AND assures a minimum amount of radioactivity plates out on turbine building surfaces.

ANSWER: 70

b. preserves personnel accessibility AND assures that radioactivity releases pass through a monitored release point.

Explanation: Continued personnel access to the turbine building, radwaste and augmented radwaste may be essential for responding to emergencies. These structures are not air tight and radioactivity release inside them would not only limit personnel access, but would eventually lead to an unmonitored ground level release. Operation of ventilation in these structures preserves accessibility, and assures that radioactivity is discharged through an elevated, monitored release point.

<u>Answer source:</u> INT008-06-17 p. 12, section B.1

Distractors:

a. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability.

c. The purpose of restarting Turbine Building ventilation is not to preserve equipment availability nor to minimize deposition of radioactivity in the building.

d. The purpose of restarting Turbine Building ventilation is not to minimize deposition of radioactivity in the building.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	3	
K/A #	2.3.4	
Importance Rating	3.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Rad Procedure 9.ALARA.1
39

Learning Objective: See Attached (As available)

Question Source:
Bank # 23484
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam CNS 2006

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

10 CFR 55 Content
55.41 12
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
71 23484	00	06/24/2006		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Matching	N

Topic Area	Description
Administrative	INT0320115, Knowledge of radiation exposure limits (ILT 2006 NRC EXAM)

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115D0D010I	Discuss the following as described in Rad Protection Procedure 9.ALARA.1, Personnel Dosimetry and Occupational Radiation Exposure Program: Lifetime TEDE Guideline

Related References	
10CFR55.41	(B)(12)

Related Skills (K/A)		ROI	SROI
2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10) (3.2/3.7) (2.5 / 3.1)		

QUESTION:

If you have exceeded your Lifetime TEDE Guideline how much exposure are you allowed during the year at CNS? What authority, if any, may grant extension to this allowed exposure?

a. 1000 mrem
Radiological Manager and Site Vice President may authorize an extension.

b. 0 mrem
No extensions are allowed.

c. 1000 mrem
No Extensions are allowed.

d. 0 mrem
Radiological Manager and Site Vice President may authorize an extension.

ANSWER:

c. 1000 mrem
No Extensions are allowed.

Explanation: The Lifetime TEDE Guideline states that NPPD shall normally limit an individual's lifetime TEDE in rem to the individual's age in years. In addition an individual exceeding the lifetime TEDE Guideline will be limited to a TEDE of 1000 mrem and will not be granted an extension.

Distractors:

a. is incorrect even though 1000 mrem are allowed no extension to this dose is allowed.

b. is incorrect because 1000 mrem TEDE is allowed.

d. is incorrect because 1000 mrem TEDE is allowed and now extensions are allowed.

Source: Direct Cognitive Level 1 Difficulty 3 10CFR55.41(b)12

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.43	
	Importance Rating	3.2	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR001-03-02
(Attach if not previously provided)
(including version/revision number) 20

Learning Objective: See Attached (As available)

Question Source: Bank # 1927
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 10
55.43

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
72 1927	00	08/10/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	N

Topic Area	Description
Systems	COR0010302001020H Communication System

Related Lessons
COR0010302 Communication

Related Objectives	
COR0010302001020H	State the purpose of the following major components in the Communications system: Emergency Notification System (ENS)

Related References	
COR0010302	Communication

Related Skills (K/A)		ROI	SROI
2.4.43	Knowledge of emergency communications systems and techniques. (CFR: 41.10 / 45.13) (3.2/3.8) (2.8 / 3.5)		

QUESTION:

An emergency has occurred at the station that requires notification of the NRC Operations Center. Which communications link is used to make this notification?

a. Emergency Notification System (ENS)

b. Health Physics Network (HPN)

c. National Warning System (NAWAS)

d. CNS State Notification System (SNS)

ANSWER:

a. Emergency Notification System (ENS)

Explanation:

In accordance with Communications Text (COR0010302); The ENS is intended as the primary means of reporting emergencies and other significant events at the station to the NRC.

When a station emergency occurs, the ENS becomes the dedicated and continuous line to the NRC for the transmission of operational data.

ENS designated telephones are located in the Control Room, NRC Resident Inspector's Office Technical Support Center, and Emergency Operations Facility.

Distractors:

b. The Health Physics Network (HPN) is not part of the ENS.

c. The National Warning System (NAWAS) is not part of the EAS.

d. The CNS State Notification System (SNS) is not part of the EAS.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.22	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): EOP 1A
15

Learning Objective: See Attached (As available)

Question Source:
Bank # 8933
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 7
55.43 ______

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	8933	02	02/01/2005		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080609, Following a LOCA with HPCI as only injection source, what is required?

Related Lessons	
INT0080609	OPS EOP FLOWCHART 1A - RPV CONTROL, RPV LEVEL

Related Objectives	
INT00806090011100	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Related References	
INT0080609	Flowchart 1A RPV Level

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: 73 8933 (1 point(s))

Following a Loss of Coolant Accident, the following conditions exist:

- All control rods inserted to or beyond position 02.
- Reactor Pressure 445 psig
- Reactor water level -185 inches corrected FZ (lowering)
- SRVs All closed
- HPCI is the **ONLY** injection source available (injecting at rated flow).

What action is now required?

a. RPV Flooding

b. Steam Cooling

c. Containment Flooding

d. Emergency Depressurization

ANSWER: 73 8933

d. Emergency Depressurization

Explanation:

With Reactor Pressure greater than the shutoff head of the low pressure injection systems and level is -185 and lowering, the reactor must be depressurized to inject with low pressure injection systems to restore reactor water level.

Distractors:

a. RPV Flooding is not required due to RPV level is known.

b. Steam Cooling is not required due to a means of RPV injection is available.

c. Containment Flooding is not required until ED is accomplished.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.32	
	Importance Rating	3.6	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 2.4ANN
(Attach if not previously provided)
(including version/revision number) 8

Learning Objective: See Attached (As available)

Question Source:

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

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	L
Comprehension or Analysis	

10 CFR 55 Content

55.41	10
55.43	

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74		00	01/08/2011		NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency	Loss of all Annunciators

Related Lessons	
INT0320136	OPS-CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT0320136M0M0100	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).

Related References
2.4ANN

Related Skills (K/A)	
2.4.32	Knowledge of operator response to loss of all annunciators. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.6 SRO 4.0

QUESTION: 74

The plant is operating at 100%. The following indications are observed:

- A Transient occurs that should have caused multiple alarms on all of the Annunciator panels.
- No alarms windows light or sound.
- No CRTs display any alarms.
- No printers print any alarms.

How should the Operator respond to this loss?

Enter Abnormal Procedure 2.4ANN…

a. and test all of the annunciator panels to determine the extent of the failure only.

b. stop all test, and power changes in progress and test all of the annunciator panels to determine the extent of the failure only.

c. stop all test, evolutions, and power changes in progress and test all of the annunciator panels to determine the extent of the failure and send Station Operators out in the field to monitor key parameters.

d. stop all test, evolutions, and power changes in progress and test all of the annunciator panels to determine the extent of the failure and send Station Operators out in the field to monitor key parameters and tell the SRO to evaluate the Emergency Action Levels for entry conditions.

ANSWER: 74

d. stop all test, evolutions, and power changes in progress and test all of the annunciator panels to determine the extent of the failure and send Station Operators out in the field to monitor key parameters and tell the SRO to evaluate the Emergency Action Levels for entry conditions.

Explanation:

A loss of annunciators during a transient meets the threshold for EALs and all test, evolutions and power changes should be stopped and the extent of the problem found by testing the annunciators.

Distractors:

a. This is incomplete.

b. This is incomplete.

c. This is incomplete.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.27	
	Importance Rating	3.9	

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Lesson COR002-19-02
(Attach if not previously provided)
(including version/revision number) 20

Learning Objective: See Attached (As available)

Question Source: Bank # 2527
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 7
55.43

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	2527	01	08/20/1999		Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	

Topic Area	Description
Systems	COR0021902001010A Reactor Equipment Cooling

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives
COR0021902001010A State the purpose of the following items related to REC: REC System

Related References	
COR0021902	REACTOR EQUIPMENT COOLING

Related Skills (K/A)
2.1.27 Knowledge of system purpose and/or function. (CFR: 41.7) IMPORTANCE RO 3.9 SRO 4.0

QUESTION: 75 2527 (1 point(s))

What is the purpose of the Reactor Equipment Cooling (REC) system?

The REC system…

a. Provides cooling water to critical and non-critical contaminated or potentially contaminated components in the Reactor, Turbine, Radwaste and Augmented Radwaste buildings.

b. Provides cooling water to the critical and non-critical, contaminated or potentially contaminated components in the Turbine, Radwaste, Augmented Radwaste, and Control buildings.

c. Provides cooling water to the critical and non-critical, contaminated or potentially contaminated components in the Reactor, Turbine, Augmented Radwaste, and Control buildings.

d. Provides cooling water to the critical and non-critical, contaminated or potentially contaminated components in the Reactor, Radwaste, Augmented Radwaste, and Control buildings.

ANSWER: 75 2527

d. Provides cooling water to the critical and non-critical, contaminated or potentially contaminated components in the Reactor, Radwaste, Augmented Radwaste, and Control buildings.

Explanation:

From Lesson COR002-19-02
SYSTEM BRIEF DESCRIPTION
A. System Purpose

The REACTOR EQUIPMENT COOLING (REC) system provides cooling water to both the critical and non-critical, contaminated or potentially contaminated components located in the Reactor, Radwaste, Augmented Radwaste, and Control buildings.

Distractors:

a. Is incorrect it has the Turbine Building listed

b. Is incorrect it has the Turbine Building listed

c. Is incorrect it has the Turbine Building listed

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #	295004.EA2.03	
Importance Rating		3.6

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Emergency Procedure 5.3DC125 23

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 1		00	1/12/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320131, CNS Abnormal Procedures (RO) Electrical

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	
5.3DC125	Loss of 125 VDC

Related Skills (K/A)	
295004.AA2.01	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) Cause of partial or complete loss of D.C. power (3.2 / 3.6)

QUESTION: S1

The plant is operating at 100% rated power when the following annunciators are received:

- 9-3-2/G-1, HPCI LOGIC POWER FAILURE.
- 9-3-2/G-2, HPCI INVERTER CKT FAILURE.

Which one of the following Abnormal/Emergency Procedures is required to be entered and what bus has failed?

a. 5.3AC120, LOSS OF 120 VAC; CDP-1A

b. 5.3AC120, LOSS OF 120 VAC; CDP-1B

c. 5.3DC125, LOSS OF 125 VDC; 125 VDC DISTRIBUTION PANEL A

d. 5.3DC125, LOSS OF 125 VDC; 125 VDC DISTRIBUTION PANEL B

ANSWER: S1

d. 5.3DC125, LOSS OF 125 VDC; 125 VDC DISTRIBUTION PANEL B

Explanation:

These two annunciators are indicative of a loss of 125 VDC Distribution Panel B and therefore an entry condition for 5.3DC125.

Distractors:

a. HPCI is a DIV II system and most of the power for it comes from DIV II 125 VDC power panels, not 120 Volt AC panels.

b. HPCI is a DIV II system and most of the power for it comes from DIV II 125 VDC power panels, not 120 Volt AC panels.

c. HPCI is a DIV II system and most of the power for it comes from DIV II 125 VDC power panels, not 125 VDC DISTRIBUTION PANEL A which is DIV I.

SRO Justification: 10CFR55.43 b. (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023.AA2.05	
	Importance Rating		4.6

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 5.7.1 Emergency Classification
43

Learning Objective: See Attached (As available)

Question Source:
Bank # 19335
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 2	19335	04	06/24/2010		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	

Topic Area	Description
Emergency Plan	GEN0030402, GEN0030401, SRO classify a dropped bundle per 5.7.1

Related Lessons	
GEN0030401	Emergency Plan for Licensed Operators
MCR0010101	On-Shift Emergency Director
GEN0030402	OPS EAL Training Part 1, Category A

Related Objectives	
GEN0030401C0C050E	Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency classification.
GEN0030402001050E	Concerning event classification: Given a copy of EPIP 5.7.1 and hypothetical abnormal plant symptoms, indications, or events, determine any and all EALs which have been exceeded and specify the appropriate emergency classification.

Related References	
5.7.1	Emergency Classification

Related Skills (K/A)	
295023.AA2.05	Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: (CFR: 41.10 / 43.5 / 45.13) ?Entry conditions of emergency plan (3.2/4.6*)

QUESTION: S2 19335 (1 point(s))

The plant is in MODE 5 with the following conditions:

- ➢ Refueling activities are in progress.
- ➢ RA-1 Refueling Floor ARM is reading 3×10^2 mRem/hr.

While transferring a fuel bundle from the Spent Fuel Pool to the Vessel the following occur:

- ➢ An irradiated fuel bundle is dropped in the cattle chute.
- ➢ REFUEL AREA HIGH RAD, 9-3-1/A-10 is in alarm (both Ronan 1448 and 1449).
- ➢ RA-1 Refueling Floor ARM is reading 5.5×10^4 mRem/hr.

What is the **MINIMUM** required Emergency Classification for this event?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

ANSWER: S2 19335

b. Alert

An ALERT should be declared per EAL AA2.1.

a. is incorrect. Conditions are met for an ALERT.

c. and d. are incorrect. Only a single bundle has been dropped. Elevation to a higher level above an ALERT would require major fuel damage defined as more than ten irradiated fuel bundles.

Reference: 5.7.1 EAL AA2.1

SRO Justification: 10CFR55.43.b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

ATTACHMENT 1 EAL AND FISSION PRODUCT BARRIER MATRICES

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
A Abnorm. Rad Release / Rad Effluent	1 Offsite Rad Conditions	AG1.1 [1 2 3 4 5 DEF] Any valid gaseous monitor reading > Table A-1 column "GE" for ≥ 15 min. (Note 1)	AS1.1 [1 2 3 4 5 DEF] Any valid gaseous monitor reading > Table A-1 column "SAE" for ≥ 15 min. (Note 1)	AA1.1 [1 2 3 4 5 DEF] Any valid gaseous monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)	AU1.1 [1 2 3 4 5 DEF] Any valid gaseous monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)
		AG1.2 [1 2 3 4 5 DEF] Dose assessment using actual meteorology indicates doses > 1 Rem TEDE or > 5 Rem thyroid CDE at or beyond the site boundary	AS1.2 [1 2 3 4 5 DEF] Dose assessment using actual meteorology indicates doses > 0.1 Rem TEDE or > 0.5 Rem thyroid CDE at or beyond the site boundary	AA1.2 [1 2 3 4 5 DEF] Any valid liquid effluent monitor reading > Table A-1 column "Alert" for ≥ 15 min. (Note 2)	AU1.2 [1 2 3 4 5 DEF] Any valid liquid effluent monitor reading > Table A-1 column "UE" for ≥ 60 min. (Note 2)
		AG1.3 [1 2 3 4 5 DEF] Field survey results indicate closed window dose rates > 1 Rem/hr expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1) OR Analyses of field survey samples indicate thyroid CDE > 5 Rem for 1 hr of inhalation at or beyond the site boundary	AS1.3 [1 2 3 4 5 DEF] Field survey indicates closed window dose rate > 0.1 Rem/hr that is expected to continue for ≥ 60 min. at or beyond the site boundary (Note 1) OR Field survey sample analysis indicates thyroid CDE > 0.5 Rem for 1 hr of inhalation at or beyond the site boundary	AA1.3 [1 2 3 4 5 DEF] Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODAM limits for ≥ 15 min. (Note 2)	AU1.3 [1 2 3 4 5 DEF] Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODAM limits for ≥ 60 min. (Note 2)
	2 Onsite Rad Conditions & Spent Fuel Pool Events	(see Table A-1 below)		AA2.1 [1 2 3 4 5 DEF] Damage to irradiated fuel OR loss of water level (uncovering irradiated fuel outside the RPV) that causes EITHER of the following: Valid RMA-RA-1 Fuel Pool Area Rad reading > 50 R/hr OR Valid RMP-RM-452 A-D Rx Bldg Vent Exhaust Plenum Hi-Hi alarm	AU2.1 [1 2 3 4 5 DEF] Unplanned water level drop in the reactor cavity or spent fuel pool as indicated by any of the following: • LI-00 (calibrated to 1001' elev.) • Spent fuel pool low level alarm • Visual observation AND Valid area radiation monitor reading rise on RMA-RA-1 or RMA-RA-2
				AA2.2 [1 2 3 4 5 DEF] A water level drop in the reactor refueling cavity or spent fuel pool that will result in irradiated fuel becoming uncovered	AU2.2 [1 2 3 4 5 DEF] Unplanned valid area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels* * Normal levels can be considered as the highest reading in the past 24 hours excluding the current peak value
	3 MCR/CAS Rad			AA3.1 [1 2 3 4 5 DEF] Dose rates > 15 mRem/hr in EITHER of the following areas requiring continuous occupancy to maintain plant safety functions: Main Control Room (RM-RA-20) OR CAS	
E ISFSI		None	None	None	EU1.1 [N/A] Damage to a loaded cask confinement boundary
F Fission Product Barriers		FG1.1 [1 2 3] Loss of any two barriers AND Loss or potential loss of third barrier (Table F-1)	FS1.1 [1 2 3] Loss or potential loss of any two barriers (Table F-1)	FA1.1 [1 2 3] Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)	FU1.1 [1 2 3] Any loss or any potential loss of Primary Containment (Table F-1)

Table A-1 Effluent Monitor Classification Thresholds

	Monitor	GE for ≥ 15 min.	SAE for ≥ 15 min.	ALERT for ≥ 15 min.	UE for ≥ 60 min.
GASEOUS	ERP	3.50E+08 µCi/sec	3.50E+07 µCi/sec	2.90E+06 µCi/sec	2.24E+05 µCi/sec
	Rx Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.45E+05 µCi/sec	5.48E+04 µCi/sec
	Turb Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.62E+05 µCi/sec	9.02E+04 µCi/sec
	RW / ARW Bldg Vent	3.50E+07 µCi/sec	3.50E+06 µCi/sec	5.64E+05 µCi/sec	9.08E+04 µCi/sec
LIQUID	Rad Waste Effluent	—	—	The lesser of: 200 x calculated alarm values OR monitor upscale*	The lesser of: 2 x calculated alarm values OR monitor upscale*
	Service Water Effluent	—	—	4.80E-04 µCi/cc	4.80E-06 µCi/cc

* with effluent discharge not isolated

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295018.AA2.03	
	Importance Rating		3.5

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): T.S. 3.7.3

Amendment 232

Learning Objective: See Attached (As available)

Question Source: Bank # 19227
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41
55.43 2

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 3	19227	01	03/10/2004		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	H	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070605, Both divisions of REC Inop (No LOSF)

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070508	OPS Tech. Specs. 3.7, Plant Systems
INT0070605	OPS TRM - Safety Function Determination Program and Other TRM Required Programs

Related Objectives	
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.
INT00705080010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.7 LCO.
INT00705080010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.7 LCO, determine the ACTIONS that are required.
INT00706050010300	Given plant conditions, TS and the TRM, determine if a Safety Function Determination (SDF) is required.
INT00706050010400	Given plant conditions, TS and TRM, determine if a Loss of Safety Function (LOSF) exists.

Related References	
2.0.11.1	Safety Function Determination Program
3.7.3	REC System
3.0.6	LCO Applicability

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)
295018.AA203	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / 43.5 / 45.13) Cause for partial or complete loss (3.2 / 3.5)

QUESTION: S 3 19227 (1 point(s))

The plant was operating at rated power when the following occurred:

- ➢ 9/8 at 0900, REC pump C becomes inoperable
- ➢ 10/3 at 1100, the shaft on SW-MO-886, SW SUPPLY TO NORTH CRITICAL LOOP is found sheared.

Per Technical Specifications, IF conditions do not change, must the CONDITIONS and REQUIRED ACTIONS of LCO 3.5.1 be entered AND what is the EARLIEST date and time the plant is required to enter MODE 4?

a. NO. 10/4 at 2300.

b. NO. 11/3 at 2300.

c. YES. 10/4 at 2300.

d. YES. 11/3 at 2300.

ANSWER: S 3 19227

a. NO. 10/4 at 2300.

Service water backup is required for REC operability (Bases p. B 3.7-13)

Pump C is Division II. MO-886 is Division I. Both loops of REC are inoperable, but NO loss of safety function exists as the LOSF evaluation does not assume a loss of off-site power which leaves adequate pumps. 3.5.1 does not need to be entered.

SRO Justification: 10CFR55.43. b (2) Facility operating limitations in the technical specifications and their bases.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	2.2.42	
	Importance Rating		4.6

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) D3.3.2 Gaseous Effluent Monitoring
(Attach if not previously provided)
(including version/revision number) 10/10/01

Learning Objective: See Attached (As available)

Question Source: Bank # 5448
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 ____
55.43 2

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 4	5448	02	06/11/2004		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Off-Site Dose Assessment Manual	INT0070702, Offgas monitors inop with high release rates

Related Lessons	
INT0070702	ODAM Specifications
INT0070508	OPS Tech. Specs. 3.7, Plant Systems

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.
INT00705080010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.7 LCO.
INT00705080010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.7 LCO, determine the ACTIONS that are required.

Related References	
D3.3.2	Gaseous Effluent Monitoring

Related Skills (K/A)	
2.2.38	Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13) (3.6/4.5)
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2) (4.0/4.7)
2.2.40	Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3) (3.4/4.7)
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: 382 5448 (1 point(s))

The plant is at 100% power with the following conditions:

- ➢ Both channels of Offgas Radiation Monitoring are incapable of providing a trip signal.
- ➢ Rising radiation levels occur on many ARMs, Main Steam Line Radiation Monitors and Offgas radiation monitors.
- ➢ The release rate at the Steam Jet Air Ejector discharge is 2.2 Ci/sec.
- ➢ Radiation levels have been stable for the last 23 minutes.

If conditions do not improve, what are the **MOST** restrictive actions required per Technical Specifications, Technical Requirements Manual and the Offsite Dose Assessment Manual?
(NOTE: The choices are listed from LEAST restrictive to MOST restrictive.)

a. Confirm that the Elevated Release Point radiation monitor is operable **AND** take grab samples of Offgas once per 24 hours.

b. Restore gross gamma activity rate of the noble gasses to within the limits within 72 hours **AND** if not restored within the limit, isolate all Main Steam lines or isolate SJAE within 12 hours.

c. Isolate the offgas system within 30 minutes **AND** lower reactor power to < 25% within 8 hours.

d. Immediately isolate the offgas system **AND** immediately initiate a reactor shutdown **AND** be in MODE 4 within 24 hours.

ANSWER: 382 5448

d. Immediately isolate the offgas system **AND** immediately initiate a reactor shutdown **AND** be in MODE 4 within 24 hours.

DLCO 3.3.2, condition "K". Tech Spec 3.7.5 allows 72 hours to restore gross gamma activity rate to within limits.

<u>SRO Justification:</u> **(2) Facility operating limitations in the technical specifications and their bases.**
SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance.

DO NOT PROVIDE TS BASES

D 3.3 INSTRUMENTATION

D 3.3.2 Gaseous Effluent Monitoring

DLCO 3.3.2 The gaseous effluent radiation monitoring instrumentation channel(s) shown in Table D3.3.2-1 shall be OPERABLE with:

a. The minimum OPERABLE channel(s) in service.

b. The alarm and trip setpoints set to ensure that the limits of DLCO 3.2.1 are not exceeded.

APPLICABILITY: According to Table D3.3.2-1.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gaseous effluent radiation monitoring instrumentation channel alarm and trip setpoint less conservative than required.	A.1 Suspend gaseous effluent radiation release monitored by inoperable channel. <u>OR</u> A.2 Declare channel inoperable.	Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more channels inoperable.	B.1 Enter the Condition referenced in Table D3.3.2-1 for the channel.	Immediately
	AND	
	B.2.1 Restore inoperable channel(s) to OPERABLE status.	31 days
	OR	
	B.2.2 In lieu of any other report, explain in the Radioactive Effluent Release Report why the instrument was not repaired in a timely manner.	In accordance with the Radioactive Effluent Release Report frequency
C. As required by Required Action B.1 and referenced in Table D3.3.2-1.	C.1 Ensure the offgas delay system is not bypassed.	Immediately
	AND	
	C.2 Ensure the Elevated Release Point Monitoring noble gas activity monitor is OPERABLE.	Immediately
	AND	
	C.3 Restore inoperable channels to OPERABLE status.	72 hours
D. Required Action and associated Completion Time for Condition C not met.	D.1 Be in MODE 2.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table D3.3.2-1.	E.1 Estimate flowrate.	24 hours AND Once per 24 hours thereafter
F. As required by Required Action B.1 and referenced in Table D3.3.2-1.	F.1 Take grab samples. AND F.2 Analyze for gross activity.	24 hours AND Once per 24 hours thereafter 24 hours from time of sampling completion

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action B.1 and referenced in Table D3.3.2-1.	G.1.1 Verify one Function 2.a monitor OPERABLE. AND	Immediately
	G.1.2 Verify recombiner exhaust temperature change less than 10°F over a 24 hour period.	Immediately AND Once per 24 hours thereafter.
	OR	
	G.2.1 Verify one Function 2.a monitor OPERABLE. AND	Immediately
	G.2.2 Collect gas sample. AND	24 hours AND Once per 24 hours thereafter
	G.2.3 Analyze gas sample and ensure within the limit of DLCO 3.2.6.	4 hours from time of sampling completion
	OR	
	G.3.1 Verify recombiner exhaust temperature change less than 10°F over a 24 hour period. AND	Immediately AND Once per 24 hours thereafter.
	G.3.2 Collect gas sample.	24 hours AND Once per 24 hours thereafter.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
		AND	
	G.3.3	Analyze gas sample and ensure within the limit of DLCO 3.2.6.	4 hours from time of sampling completion.
H. Required Action and associated Completion Time for Condition G not met.	H.1	Discontinue operation of the augmented offgas treatment system.	Immediately
I. As required by Required Action B.1 and referenced in Table D3.3.2-1.	I.1	-------NOTE------- When the primary monitoring system is inoperable and the backup system is in service, sampling may be discontinued for up to 30 minutes only for changing particulate filters and iodine cartridges. -------------------	
		Continuously collect samples with auxiliary sampling equipment as required in Table D3.2.3-1.	4 Hours
	OR		
	I.2.1	If auxiliary sampling equipment cannot be established within the specified completion time, enter the problem into the Corrective Action Program to evaluate particulate and iodine effluent releases.	Immediately
		AND	
	I.2.2	Report this event in the Radioactive Effluent Release Report.	In accordance with the Radioactive Effluent Release Report Frequency

INFORMATION ONLY

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
J. Required Action and associated Completion Time for Condition E or F not met.	J.1 Discontinue effluent releases via this pathway.	Immediately
K. Function 1.a trip capability not maintained. AND Radiation level exceeds 1.0 ci/sec (prior to 30 min. delay line) for > 15 consecutive minutes.	K.1 Close the offgas isolation valve. AND K.2 Initiate reactor shutdown. AND K.3 Be in MODE 4.	Immediately Immediately 24 hours

Table D3.3.2-1 (Page 1 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITIONS	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Steam Jet Air Ejector				
a. Noble Gas Activity Monitor	(a)	1(e)	C	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.8 DSR 3.3.2.10 DSR 3.3.2.11 DSR 3.3.2.12
b. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
2. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor (2% monitor)	(c)	2	G	DSR 3.3.2.1 DSR 3.3.2.4 DSR 3.3.2.6
3. Reactor Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
f. Isolation Monitor	(d)	(d)	(d)	DSR 3.3.2.5 DSR 3.3.2.11

(continued)

(a) During operation of the steam jet air ejector

(b) During releases via this pathway

(c) During augmented offgas treatment system operation

(d) See Technical Specification 3.3.6.2

(e) Second channel must either be OPERABLE or be in the tripped condition.

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

Table D3.3.2-1 (Page 2 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITION	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
4. Elevated Release Point Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
5. Radwaste Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
6. Turbine Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	(b) (f)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.9 DSR 3.3.2.10
b. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2

(continued)

(b) During releases via this pathway

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

Table D3.3.2-1 (Page 3 of 3)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIAL CONDITION	MINIMUM CHANNELS OPERABLE	CONDITION REFERENCED FROM ACTION B.1	SURVEILLANCE REQUIREMENTS
6. (continued)				
c. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
d. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
e. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
7. Multi Purpose Facility (MPF) Building Ventilation Monitoring System				
a. Iodine Sampler Cartridge	(b) (f)	1	I	DSR 3.3.2.2
b. Particulate Sampler Filter	(b) (f)	1	I	DSR 3.3.2.2
c. Effluent System Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10
d. Sampler Flow Rate Measuring Device	(b)	1	E	DSR 3.3.2.1 DSR 3.3.2.7 DSR 3.3.2.10

(b) During releases via this pathway

(f) A channel may be removed from service for up to 30 minutes for changing particulate filters or iodine cartridges or for low flow alarm check without entering Conditions or Required Actions.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295024	G 2.4.6
	Importance Rating		4.7

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): EOP3A and EOP/SAG Graphs
13 14

Learning Objective: See Attached (As available)

Question Source:

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

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	H

10 CFR 55 Content

55.41	
55.43	5

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 5		00	01/10/2011		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	5	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT008-06-18, OPS EOP and SAG Graphs and Cautions

Related Lessons
INT0080618 OPS EOP and SAG Graphs and Cautions

Related Objectives	
INT00806180011200	For each graph used in the flowcharts, identify the action(s) required if the parameters associated indicate operation in the restricted or prohibited area.

Related References	
5.8	Emergency Operating Procedures (EOPs)
EOP SAG Graphs	

Related Skills (K/A)	
295024	High Drywell Pressure
2.4.6	Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.7 SRO 4.7

QUESTION: S5

During a LOCA the following conditions exist:

- Average drywell temperature is 210°F and steady.
- Average suppression pool temperature is 205°F and rising.
- Drywell maximum run temperature is 200°F and steady.
- Torus water level is 14 feet and rising.
- Reactor pressure is 1000 psig and steady.
- Reactor water level +25 inches (narrow range)

<u>IF</u> the mitigating strategy for these conditions were to Emergency RPV Depressurization, what potential consequence could result?

a. Reactor water level indication may be lost.

b. The Safety Relief Valve (SRV) tail pipe supports may fail.

c. The Primary Containment Pressure Limit (PCPL) may be exceeded.

d. The Torus-to-Drywell Vacuum Breaker capacity may be exceeded.

ANSWER: S5

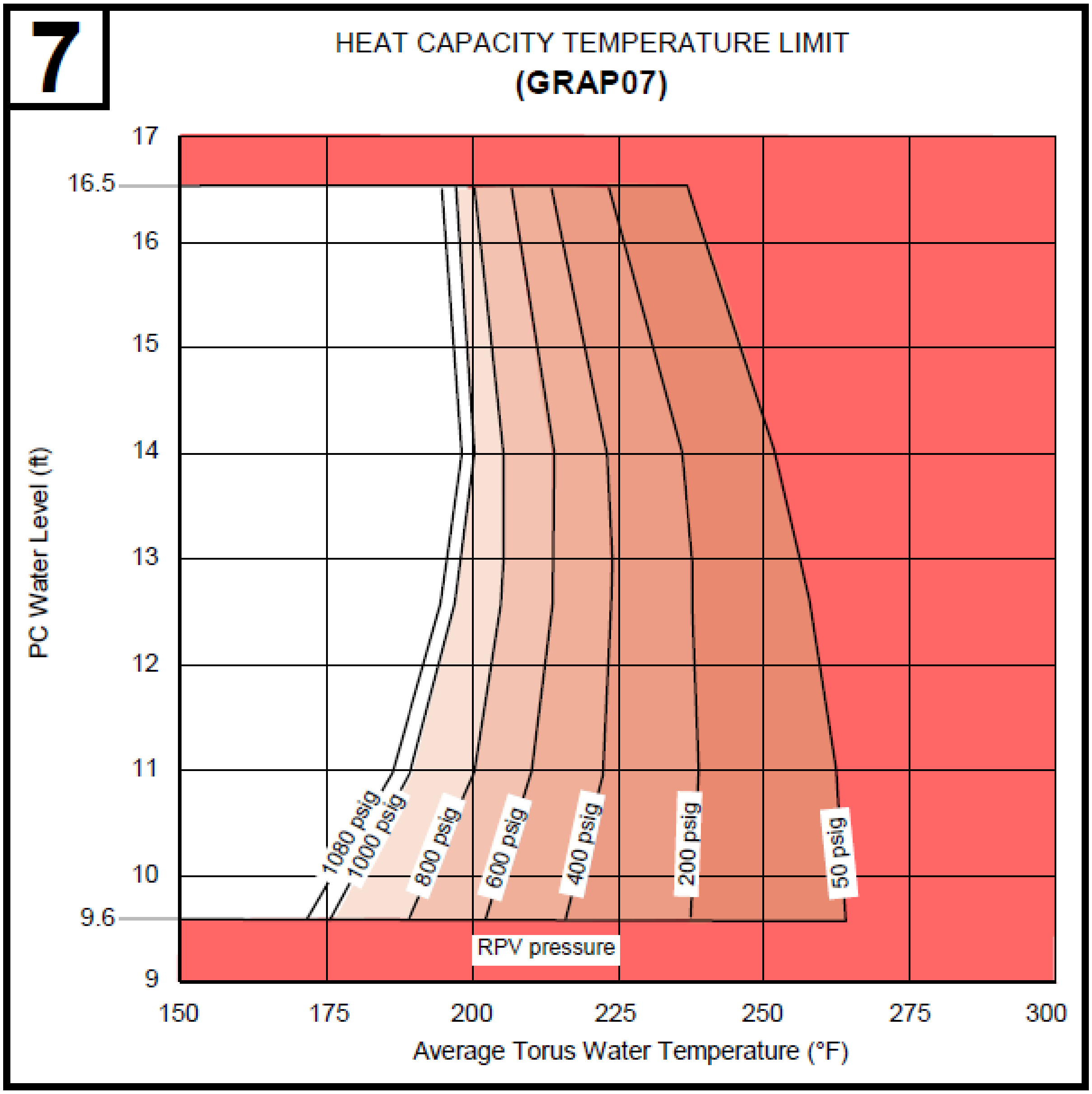
c. The Primary Containment Pressure Limit (PCPL) may be exceeded.

PROVIDE THE STUDENTS WITH EOP/SAG GRAPH 7

Current conditions place the plant on the unsafe side of the HCTL Curve. An ADS initiation now may result in exceeding the PCPL due to insufficient energy absorption capacity to handle a blowdown.

a. is incorrect 200°F is below the saturation limit curve.
b. is incorrect The maximum level where a blowdown would not cause damage to the SRV tail pipe supports is 16 feet.
d. is incorrect Torus to Drywell Vacuum Breakers are designed for LOCA energy release.

SRO Justification: 10CFR55.43. b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.


7
HEAT CAPACITY TEMPERATURE LIMIT
(GRAP07)
17
16.5
16
15
14
13
12
11
10
9.6
9
PC Water Level (ft)
1080 psig
1000 psig
800 psig
600 psig
400 psig
200 psig
50 psig
RPV pressure
150
175
200
225
250
275
300
Average Torus Water Temperature (°F)

PC-2

PC
PRESSURE

PC/P-1

Monitor and control PC pressure below 1.84 psig using containment pressure control systems, EOP 5.8.17

PC/P-2

BEFORE

torus pressure reaches 10 psig

TORUS SPRAY IS REQUIRED

8

PC/P-3

WHEN

torus pressure exceeds 10 psig

DRYWELL SPRAY IS REQUIRED

PC/P-4

WHEN

torus pressure cannot be maintained below PSP (GRAPH 10)

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

PC/P-5

BEFORE

drywell pressure reaches PCPL-A (GRAPH 11)

PC/P-6

Emergency vent PC, EOP 5.8.18, to control pressure below PCPL-A (GRAPH 11)
(exceed offsite radioactivity release rate limits if necessary)
(defeat isolation interlocks if necessary)

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295027 G 2.1.39	
	Importance Rating		4.3

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 2.0.3 Conduct of Operations
72

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 6		00	1/10/2011		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	

Topic Area	Description
Administrative	

Related Lessons
INT0320103 CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives
INT032010300C010G Discuss the following as described in conduct of Operations Procedure 2.0.3, Conduct of Operations: Operations Policy During Transient Operations

Related References
2.0.3 Conduct of Operations

Related Skills (K/A)
295027 High Containment Temperature 2.1.39 Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12) IMPORTANCE RO 3.6 SRO 4.3

QUESTION: S6

The plant is at 100% power with rising Containment temperatures due to multiple Drywell Fan Cooling Units.

As Primary Containment temperature approaches the Tech Spec Limit the SRO directs the RO to rapidly reduce reactor power in accordance with Procedure 2.1.10 to reduce the heat load in containment.

What is the Tech Spec Limit for Average Drywell Temperature, and in which procedure would the SRO find the guidance to take these conservative actions?

a. 135°F, and COP 2.0.3 (CONDUCT OF OPERATIONS)

b. 135°F, and COP 2.0.1 (PLANT OPERATIONS POLICY)

c. 150°F, and COP 2.0.3 (CONDUCT OF OPERATIONS)

d. 150°F, and COP 2.0.1 (PLANT OPERATIONS POLICY)

ANSWER: S6

c. 150°F, and COP 2.0.3 (CONDUCT OF OPERATIONS)

Explanation:

The Tech Spec Limit for Average Drywell Temperature is 150°F, in accordance with T.S. 3.6.1.5 Drywell Air Temperature. Conduct of Operations Procedure 2.0.3 contains the guidance for taking conservative actions on safety and non-safety systems in section 8.2.

Distractors:

a. 135° is the normal operating temperature not the tech spec limit.

b. 135° is the normal operating temperature not the tech spec limit. COP 2.0.1 does not contain the guidance for conservative decision making.

d. COP 2.0.1 does not contain the guidance for conservative decision making

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Drywell Air Temperature

LCO 3.6.1.5 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.5.1 Verify drywell average air temperature is within limit.	24 hours

RESPONSIBILITY AND AUTHORITY FOR THE OPERATION OF CNS

1.1.1 VICE PRESIDENT-NUCLEAR AND CHIEF NUCLEAR OFFICER

1.1.1.1 Reports directly to the President and Chief Executive Officer of NPPD.

1.1.1.2 Has corporate responsibility for overall plant nuclear safety and taking measures, as needed, to ensure acceptable performance of the Staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

1.1.2 GENERAL MANAGER OF PLANT OPERATIONS

1.1.2.1 Reports directly to the Vice President-Nuclear and Chief Nuclear Officer.

1.1.2.2 Overviews and coordinates responsibility for ensuring safe and efficient operation of Cooper Nuclear Station.

1.1.2.3 Nuclear Operations is headed by the General Manager of Plant Operations (GMPO), who is responsible for the safe, reliable, and efficient operation of CNS. This responsibility includes: Operations; Maintenance; Radiation Protection; Chemistry; Planning, Scheduling, and Outages; Industrial Safety; Quality Control; Human Performance; and those on-site activities necessary for safe operation and maintenance of the plant.

1.1.2.4 Responsible for ensuring CNS is operated in compliance with the Operating License, Technical Specifications, and all applicable Federal and State regulations.Ⓟ[9]

1.1.2.5 Overviews and coordinates the activities of the following positions: Operations Manager; Maintenance Manager; Chemistry Manager; Radiation Protection Manager; Planning, Scheduling, and Outages Managers; Industrial Safety Coordinator; Quality Control Coordinator; and Human Performance Coordinator.

1.1.3 OPERATIONS MANAGER

1.1.3.1 The Operations Manager reports directly to the General Manager of Plant Operations and is responsible for assisting him in the safe, reliable, and efficient operation of CNS. The Operations Manager has overall responsibility for startup, shutdown, refueling operations, day-to-day operation of the plant, and Work Control Center activities.

1.1.4 ASSISTANT OPERATION'S MANAGER OF SHIFT SUPPORT (AOM-SHIFT)

1.1.4.1 The AOM-Shift reports to the Operations Manager and is directly responsible for fuel loading, startup, operation, and shutdown of all station equipment associated with operation of the power plant.

1.1.4.2 The AOM-Shift shall ensure Operating Crews apply a conservative operating philosophy, fulfill their roles and responsibilities regarding reactivity management, and adhere to reactivity management standards established in this procedure.Ⓟ[4]

1.1.4.3 The AOM-Shift shall hold an SRO License on the facility.Ⓟ[1]

1.2 SHIFT MANAGER (SM)

1.2.1 The Shift Manager (SM) is the Senior Licensed Operator on-watch and is at all times the Operations and General Manager of Plant Operations direct representative for the conduct of operations at CNS. The SM has the responsibility and authority to direct all activities and personnel at CNS, as required to:

1.2.1.1 Protect the health and safety of the public and the environment.

1.2.1.2 Ensure all plant evolutions are performed by an approved station document.

1.2.1.3 Ensure all plant evolutions are in compliance with the Radiation Protection Program.

1.2.1.4 Prevent damage to equipment and structures.

1.2.1.5 Protect the physical security of CNS.

1.2.1.6 Ensure compliance with Technical Specifications, Operating License, and the USAR.Ⓟ[9]

1.2.1.7 The SM shall maintain overall responsibility for control of the key shutdown safety functions inclusive of opening and closing the associated system work windows.Ⓟ[28]

1.2.2 On holidays, weekends, and backshifts when the Station is at a minimum staffing level, the SM is the Senior Management representative on-site.

1.2.3 The SM's authority to act to ensure compliance with licensing requirements and to ensure the safe operation of the unit or equipment shall overrule all other Management authority.

1.2.4 The SM shall hold an active Senior Reactor Operator License and will report directly to the AOM-Shift.Ⓟ[1]

1.2.5 SM general management responsibilities include:

1.2.5.1 The SM maintains an oversight role cognizant of all plant evolutions occurring on the shift.

a. The SM shall maintain oversight and control of operations, maintenance, and surveillance activities during all modes of plant operations through the CRS, WCO, STE, WCCA, and other work organization Supervisors.

b. The SM shall provide an oversight function to the CRS, WCO, and the STE during normal, abnormal, and emergency operations.

c. The SM should refrain from manipulating equipment.

d. The SM shall provide guidance and assistance, when necessary, to the CRS in abnormal and emergency conditions.

1.2.5.2 The SM shall ensure conservative operating practices are followed and that safety and core integrity take precedence over power production. This includes:Ⓟ[4,6,7]

a. Ensuring Operations personnel control reactivity and take conservative action to safeguard integrity of the reactor fuel.

b. Ensure hasty decisions and hurried actions are avoided whenever possible, during reactivity manipulations.

c. Ensure adding positive reactivity during abnormal or transient events as a means to mitigate the event is avoided whenever possible.

1.2.5.3 The SM is an active participant in the Work Control process, ensuring the proper focus on identified equipment deficiencies are prioritized correctly, ensuring safe and reliable operation of the facility.Ⓟ[6]

a. The SM shall be cognizant of the work identified in the Plan of the Day, including the extent necessary to fully understand actual and potential impacts to the reactor core and associated support systems, and therefore, ensure the identified work can be performed under existing plant conditions.

   1. The SM shall also be cognizant of the actual and potential impacts upon the Risk Significant and Shutdown Safety Functions made within the schedule.Ⓟ[28]

b. The SM should communicate to the CRS, WCO, STE, WCCA, and Operations Shift the activities that are planned for the shift and the priority of the activities.

c. The SM should ensure surveillance scheduled for Operations are performed before expiration of the grace period and ensure required surveillance are current before changing plant operating conditions.

1.2.5.4 The SM shall ensure adherence with Operation's standards and expectations, focusing on the Operating Crew.Ⓟ[6]

a. The SM shall promote and define a continuous improvement philosophy within the Operations Department focusing on the Crews' performance, departments practices, policies, and processes.

b. The SM shall serve as the Crew mentor, providing the necessary guidance and coaching fostering an environment that promotes striving for excellence amongst the Crew.

1.2.5.5 The SM shall ensure the shift manning requirement is met for all modes of operation, as defined in this procedure.

a. The SM should ensure the on-shift Staff is adequate to perform the work scheduled for the shift and ensure sufficient manpower is available for the activities planned for the next shift.

1.2.5.6 The SM should conduct a pre-shift brief with the on-coming Crew, encompassing, but not limited to plant conditions, planned evolutions, and any pertinent plant information as required.

1.2.5.7 The SM should perform end of shift panel walkdowns that focus on safety-related controls manipulated during the shift. This walkdown may be performed as part of shift turnover.

1.2.5.8 The SM may assist the Crew in performing hourly panel walkdowns as described in Step **Error! Reference source not found.**.

1.2.5.9 The SM shall evaluate plant events for immediate reportability and Technical Specifications compliance and ensure initial notifications are made in accordance with applicable procedural guidance.Ⓟ[9]

a. If circumstances permit, the SM should contact the Operations Manager or designee prior to making any immediate reports.

1.2.5.10 The SM is responsible to recognize and prioritize all emergent issues and contacting Senior Management and/or other plant disciplines, as necessary.

a. Ensures Doniphan Control Center is apprised of impending CNS plant conditions that could result in a power reduction or shutdown of CNS within the ensuing 24 hours.Ⓟ[11]

1.2.5.11 The SM shall assume the position of Emergency Director until relieved in accordance with applicable Emergency Plan Implementing Procedures.

1.2.5.12 The SM is the Operations Department Manager's delegate responsible for closure for all NAIT items assigned to the Crew.

1.2.5.13 The SM shall be available to report to the Control Room within 10 minutes at all times.

1.2.6 SM responsibilities specific to reactivity management include:Ⓟ[6]

1.2.6.1 The SM shall notify Management per Operation Instruction #12 of reactivity management related events as described in Attachment 1 of Procedure 0-CNS-61.Ⓟ[6]

1.2.6.2 The SM shall ensure all requirements and expectations associated with standards for reactivity manipulation with control rods are carried out by Operations and Reactor Engineering Department personnel.Ⓟ[6]

1.2.6.3 The SM shall establish himself as the highest authority for controlling the reactor core.Ⓟ[6]

1.2.6.4 The SM shall review and approve all recommendations from Reactor Engineering for control rod manipulation prior to performance.Ⓟ[6]

1.2.6.5 The SM shall ensure a Reactivity Manager is assigned to provide dedicated oversight to the reactivity manipulation per Section **Error! Reference source not found.**.

NOTE **–** Step **Error! Reference source not found.** defines a significant reactivity manipulation or power changes referred to in this procedure.

1.2.6.6 The SM shall ensure all necessary support personnel are on-site for support of shift operations for significant reactivity manipulation or power changes evolutions prior to performing the activity.

1.2.6.7 The SM shall ensure a pre-activity brief, lead by the Reactivity Manager, is conducted with all involved Operations and Reactor Engineering personnel prior to performing significant reactivity manipulations or power changes.Ⓟ[6,7]

a. The SM is also responsible to ensure all personnel are properly trained and qualified to perform their assigned tasks.

1.2.6.8 The SM shall ensure all activities that are Operator distractions are suspended, including restricting access to Control Room, during performance of significant reactivity manipulations or power changes.Ⓟ[7]

a. This shall include reactivity changes that occur during shift change or Crew briefings unless all reasonable efforts are made to ensure absolute minimum distractions to the Staff performing the operation.

1.2.6.9 The SM shall enforce strict standards for all planned reactivity changes including expectations for use of peer checks, human error prevention methods (e.g., STAR), and adherence to the communication standard.Ⓟ[6]

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	G 2.4.3
	Importance Rating		3.9

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) T.S. 3.3.3.1 Post accident monitoring (PAM) instrumentation

(Attach if not previously provided)
(including version/revision number) Amendment 178

Learning Objective: See Attached (As available)

Question Source:			
	Bank #	19967	
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam NA

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

10 CFR 55 Content		
	55.41	
	55.43	2

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 7	19967	01	03/20/2003		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070504, How long to MODE 3 with PAM instruments inop

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	
3.3.3.1	Post accident monitoring (PAM) instrumentation

Related Skills (K/A)	
216000.K3.25	Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following: (CFR: 41.7 / 45.4) Vessel pressure monitoring (3.9/4.1)
2.4.3	Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4) IMPORTANCE RO 3.7 SRO 3.9

QUESTION: S7 19967 (1 point(s))

The plant is operating at rated power with the following conditions:

- NBI-PR-2A (Wide Range Reactor Pressure) becomes inoperable on 3/28 at 1100.
- NBI-LI-85A (Wide Range RPV water level) becomes inoperable on 3/29 at 0900.
- NBI-PR-2B (Wide Range Reactor Pressure) becomes inoperable on 4/1 at 1500.

IF conditions do not change, when are you ***required*** to enter **MODE 3** by Technical Specifications?

a. 1500 on 4/8

b. 0300 on 4/9

c. 1500 on 5/1

d. 0300 on 5/2

ANSWER: S7 19967

b. 0300 on 4/9

Enter 3.3.3.1.A and 3.3.1.C at 1500 on 4/1. Enter 3.3.3.1.D at 1500 on 4/8. Enter 3.3.3.1.E at 1500 on 4/8. Be in MODE 3 by 0300 on 4/9.

Answer source: LCO 3.3.3.1

Distractors:

a. Not adding the 12 hours.
c. Not adding 12 hours and not entering condition C and misreading A.
d. not entering condition C and misreading A.

Provide to Candidate: T.S. LCO 3.3.3.1 and bases.

2004 Biennial Exam

SRO Justification: 10CFR55.43. b (2) Facility operating limitations in the technical specifications and their bases.

3.3 INSTRUMENTATION

3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3.1 The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function. For Function 5, separate Condition entry is allowed for each penetration flow path.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately
C. One or more Functions with two required channels inoperable. <u>OR</u> One Function 2.c channel inoperable.	C.1 Restore one required channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Enter the Condition referenced in Table 3.3.3.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	E.1 Be in MODE 3.	12 hours
F. As required by Required Action D.1 and referenced in Table 3.3.3.1-1.	F.1 Initiate action in accordance with Specification 5.6.6.	Immediately

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Pressure	2	E
2. Reactor Vessel Water Level		
a. Fuel Zone	2	E
b. Wide Range	2	E
c. Steam Nozzle	1	F
3. Suppression Pool Level (Wide Range)	2	E
4. Primary Containment Gross Radiation Monitors	2	F
5. PCIV Position	2 per penetration flow path[(a)(b)]	E
6. Primary Containment H_2 & O_2 Analyzer	2	E
7. Primary Containment Pressure		
a. Drywell Narrow Range	2	E
b. Drywell Wide Range	2	E
c. Suppression Chamber Wide Range	2	E
8. Suppression Pool Water Temperature	2[(c)]	E

(a) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel requires a minimum of four resistance temperature detectors (RTDs) to be OPERABLE with no two adjacent RTDs inoperable.

INFORMATION ONLY

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295033.EA2.01	
	Importance Rating		3.9

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) Emergency Procedure 5.7.1
(Attach if not previously provided)
(including version/revision number) 43

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41
55.43 5

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 8		00	01/10/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	GEN0030403, Determine the highest EAL classification required for C S/D

Related Lessons	
GEN0030401	Emergency Plan for Licensed Operators
GEN0030403	OPS EAL Training Part 2, Category S and Category C

Related Objectives	
GEN0030401C0C050A	Concerning event classification: Given a copy of EPIP 5.7.1 and an EAL identification code, determine the EAL and its associated emergency classification.
GEN0030403001050A	Concerning event classification: Given a copy of EPIP 5.7.1 and an EAL identification code, determine the EAL and its associated emergency classification.

Related References	
5.7.1	Emergency Classification

Related Skills (K/A)	
295033.EA2.01	Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: (CFR: 41.10 / 43.5 / 45.13) Area radiation levels (3.8 / 3.9)

QUESTION: S8

The Reactor is in Cold SHUTDOWN when the following events occur:

- Normal Area Radiation Monitor readings for the NE Quad is 5 mRem/hr., SE Quad is 4 mRem/hr., NW Quad is 3 mRem/hr., SW Quad is 6 mRem/hr.

1 hour later, the following conditions exist:

- All vital 125 VDC buses indicate 100 Volts DC and have for the last 20 minutes.
- Two area Reactor Building Sumps have started pumping much more frequently.
- The NW Quad radiation levels are steady at 4250 mRem/hr.
- The SE Quad radiation levels are steady at 3750 mRem/hr.

Determine all applicable EAL classifications required for these conditions?

a. SAE per SS7.1 and NOUE per AU2.2

b. SAE per SS7.1 and NOUE per HU1.4

c. NOUE per CU6.1 and NOUE per AU2.2

d. NOUE per CU6.1 and NOUE per HU1.4

ANSWER: S8

c. NOUE per CU6.1 and NOUE per AU2.2

Explanation:

With the Reactor in Cold Shutdown the EAL should be based on Cold S/D or A all modes. In this case, the radiation levels have risen >1000 times normal in the NW Quad requiring the SRO to assess EALs A2 Onsite Rad Conditions and in the UE column under AU2.2 Unplanned vital area radiation monitor reading or survey results rise by a factor of 1,000 over normal levels is required to be declared. Also with 125 VDC voltage at 100 Volts for greater than 15 minutes, a NOUE should be declared under CU6.1 cold shutdown and loss of DC power.

Distractors:

a. If the plant had been in either modes 1,2, or 3 a SAE per SS7.1 would be required to be declared.

b. If the plant had been in either modes 1,2, or 3 a SAE per SS7.1 would be required to be declared. The leakage into the sumps is not bad enough to declare a flooding problem in accordance with HU1.4.

d. The leakage into the sumps is not bad enough to declare a flooding problem in accordance with HU1.4.

REQUIRED REFERENCES: Procedure 5.7.1

SRO Justification: 10CFR55.43.b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295020	G 2.2.22
	Importance Rating		4.7

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): T.S. 3.3.6.1

Amendment 178

Learning Objective: See Attached (As available)

Question Source:

Bank #	110	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	H

10 CFR 55 Content

55.41	
55.43	2

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 9	110	0	04/01/1998		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	1	Multiple Choice	

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	
INT00705040010100	Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Related References	
3.3.6.1	Primary Containment Isolation Instrumentation

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

QUESTION: S9 110 (1 point(s))

At 1335 on 1/7 it is discovered that HPCI Temperature Elements HPCI-TS 101 A through D, HPCI-TS 102 A through D, HPCI-TS 103 A through D, and HPCI-TS 104 A through D (located in RHR Injection Valve room and Torus Area West) are set to trip at 205°F.

What actions (if any) are required by Technical Specifications?

a. No Technical Specification ACTIONS are required.

b. Restore HPCI isolation capability by 1435 on 1/7 or isolate HPCI by 1535 on 1/7.

c. Place the channels associated with the HPCI Temperature Elements in trip by 1335 on 1/8.

d. Place only the channels associated with HPCI-TS 101 A through D and HPCI-TS 104 A through D in trip by 1335 on 1/8.

ANSWER: S9 110

b. Restore HPCI isolation capability by 1435 on 1/7 or isolate HPCI by 1535 on 1/7.

Explanation:

In accordance with Tech Specs 3.3.6.1 B. One or more functions with isolation capability not maintained. The required action is to restore isolation capability within 1 hour (14:35 on 1/7). If that is not possible then Condition C applies and required action C.1 states, Enter the Condition referenced in Table 3.3.6.1-1 for the channel. That Condition from the Table is “F” which states “islate the affected penetration flow path(s), within one hour (15:35 on 1/7)

Distractors:

a. These are Primary Containment Isolation Instruments that are set too high and are inoperable in accordance with Tech Specs 3.3.6.1.

c. This is the required actions if trip capability is maintained, which it is not.

a. This is a combination of required actions for multiple inoperable instruments, however not allowed by Tech Specs for this case.

PROVIDE STUDENT WITH TECH SPEC 3.3.6.1

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		2
K/A #	295029.EA2.01	
Importance Rating		3.9

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): T.S. 3.6.2.2 & Bases 3.6.2.2 — 178 — 0

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam: NA

Question Cognitive Level:
- Memory or Fundamental Knowledge:
- Comprehension or Analysis: H

10 CFR 55 Content:
- 55.41:
- 55.43: 2

Comments:

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
S 10		00	06/02/2010		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	Suppression Pool Level High out of spec high, what is required by TS?

Related Lessons
INT0070507 CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.6.2.2	Suppression Pool Level

Related Skills (K/A)	
295029.EA2.01	Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Suppression pool water level 3.9* / 3.9*

QUESTION: S10

The Plant is in MODE 1 with the following conditions at 09:00 on September 8th:

- Suppression Pool Level as read on PC-LI-13 is +2.5 inches
- There are no methods to reduce this level available.

When is the plant required by Tech Specs to be in MODE 4?

a. 21:00 on Sept. 8^{th}

b. 23:00 on Sept. 8^{th}

c. 21:00 on Sept. 9^{th}

d. 23:00 on Sept. 9^{th}

ANSWER: S10

d. 23:00 on Sept. 9^{th}

Explanation:

The corresponding levels indicated on the narrow range suppression pool level instrument PC-LI-13 are -2 inches and +2 inches for the Tech Spec Numbers given in LCO 3.6.2.2 of $\geq$ 12 ft 7 inches and $\leq$ 12 ft 11 inches. With level at 2.5 inches this would be above the TS Limit of 12 ft 11 inches therefore requiring entry into LCO 3.6.2.2. Since water level cannot be lowered the plant must be placed in MODE 3 in 12 hours and MODE 4 in 36 hours. There are also two hours to try to get level lowered and this time is added to the 36 total hours to be in MODE 4.

Distractors:

a. 21:00 on Sept 8^{th} is incorrect; this is the time to be in mode 3 minus the extra 2 hours.

b. 23:00 on Sept 8^{th} is incorrect; this is the time to be in mode 3.

c. 21:00 on Sept 9^{th} is incorrect; this is the time to be in mode 4 minus the extra 2 hours.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	205000.A2.09	
	Importance Rating		3.8

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.2.69.2 RHR System Shutdown Operations
72

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 11	00	1/08/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0022302, RHR Pump in SDC with LPCI initiation signal

Related Lessons	
COR0010102	AC Electrical Distribution
COR0022302	RESIDUAL HEAT REMOVAL

Related Objectives	
COR0010102001090J	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Breaker Anti-Pump features
COR0022302001080N	Predict the consequences a malfunction of the following will have on the RHR system: Suction flow path
COR0022302001150B	Given plant conditions, determine if the following should occur: RHR pump start
COR0022302001150C	Given plant conditions, determine if the following should occur: RHR pump trip
COR0022302001150D	Given plant conditions, determine if the following should occur: RHR valve reposition

Related References	
PR	2.2.69.2
55.43 b (5)	

Related Skills (K/A)		ROI	SROI
203000.A2.09	Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal condition or operations: (CFR: 41.5 / 45.6) Inadequate system flow (3.3/3.4)		

QUESTION: S 11

The plant is shutdown to repair a steam leak on an outboard MSIV with the following conditions:

- Inboard MSIVs are closed
- Reactor pressure is 40 psig
- RHR loop A has been placed in shutdown cooling with the A pump.
- A large leak develops on the reactor recirculation system inside the drywell.
- RPV water level is - 50" (indicated Fuel Zone).

What sequence of actions is required to inject with the "A" RHR Pump?

a. OPEN the RHR Pump "A" suction valve RHR-MO-13A, **then** CLOSE the Shutdown Cooling suction RHR-MO-15A, **then** flag the pump control switch to OFF and release **ONLY**; IAW SOP 2.2.69 RHR System Shutdown Operations

b. OPEN the RHR Pump "A" suction valve RHR-MO-13A, **then** CLOSE the Shutdown Cooling suction RHR-MO-15A, **then** flag the pump control switch to OFF and release **ONLY**; IAW AOP 2.4SDC Shutdown Cooling Abnormal

c. CLOSE the Shutdown Cooling suction RHR-MO-15A, **then** OPEN the RHR Pump "A" suction valve RHR-MO-13A, **then** flag the pump control switch to OFF and release **ONLY**; IAW SOP 2.2.69 RHR System Shutdown Operations

d. CLOSE the Shutdown Cooling suction RHR-MO-15A, **then** OPEN the RHR Pump "A" suction valve RHR-MO-13A, **then** flag the pump control switch to OFF and release **ONLY**; IAW AOP 2.4SDC Shutdown Cooling Abnormal

ANSWER:

d. CLOSE the Shutdown Cooling suction RHR-MO-15A, **then** OPEN the RHR Pump "A" suction valve RHR-MO-13A, **then** flag the pump control switch to OFF **ONLY**; IAW AOP 2.4SDC Shutdown Cooling Abnormal

With the RHR system aligned in SDC, both torus suctions are closed and the low level condition causes the reactor suctions 17 and 18 to close, tripping the pump. As level continues to lower, a LPCI initiation signal is generated and an auto start signal is applied to the pumps, causing them to enforce their anti-pumping relay. So the operator must supply a suction path to the A RHR pump by first closing the RHR-MO-15A and then opening the RHR-MO-13A (the valves are interlocked such that the 13 can't be opened if the 15 is open). The next action would be to flag the control switch for the RHR pump to stop to clear the anti pumping relay and the pump will automatically start and inject into the vessel.

The guidance to perform these actions are in Attachment 3 of 2.4SDC Shutdown Cooling Abnormal.

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		1
K/A #	212000.A2.19	
Importance Rating		3.9

Proposed Question: See Attached

Proposed Answer: _See Attached_______

Explanation: See Attached

Technical Reference(s)
(Attach if not previously provided)
(including version/revision number)

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 12	00	1/8/2011		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0022102, quarter scram on RPS loss

Related Lessons	
COR0022102	REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001080G	Given a specific RPS malfunction, determine the effect on any of the following: Scram air header solenoid operated valves

Related References	
PR	4.5
55.43 b (5)	

Related Skills (K/A)		ROI	SROI
295015.AK2.04	Knowledge of the interrelations between INCOMPLETE SCRAM and the following: (CFR: 41.7 / 45.8) RPS (4.0/4.1) (4.0 / 4.1)		
212000.A2.19	Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the...: (CFR: 41.5 / 45.6) Partial system activation (half-SCRAM) (3.8 / 3.9)		

QUESTION: S 12

The plant is operating at 99% power with no equipment out of service.

- I&C technicians have started a test on RPS trip system 'A' that will produce a half-scram.
- You notice one of the four RPS Scram Group lights is NOT lit on RPS trip system 'B'.
- All 4 RPS Scram Group lights for trip system 'A' extinguish.

What are the consequences of this event and what action is required?

a. One quarter of the control rods insert into the core; Enter 2.4CRD CRD Trouble and manually Scram the Reactor.

b. One half of the control rods insert into the core; Enter 2.4CRD CRD Trouble and manually Scram the Reactor

c. One quarter of the control rods insert into the core; Enter 5.3AC120 Loss of 120 VAC and manually Scram the Reactor.

d. One half of the control rods insert into the core; Enter 5.3AC120 Loss of 120 VAC and manually Scram the Reactor.

ANSWER: S 12

a. One quarter of the control rods insert into the core; Enter 2.4CRD CRD Trouble and manually Scram the Reactor.

EXPLANATION OF ANSWER: a. Correct. With one (1) white SCRAM INDICTIONS GROUP A light extinguished, approximately 1/4 of the Scram Pilot Valves for RPS "A" are de-energized. When the power supply for RPS B is transferred, all B channel Scram Pilot Valves will be de-energized. This will cause all Gr. 4 rods, approximately 1/4 of all rods, to scram.

2.4CRD contains the actions to Scram the reactor in this condition (more then one control rod drifting).

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	264000	G 2.2.36	
	Importance Rating			4.2

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) T.S. 3.8.1
(Attach if not previously provided)
(including version/revision number) 233

Learning Objective: See Attached (As available)

Question Source: Bank # 21404
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

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

10 CFR 55 Content 55.41
55.43 2

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 13 21404	01	10/15/2004	05/23/2010	Open Reference	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	6	Multiple Choice	N

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070509, SRO ONLY, TS 3.8.1 and TS 3.5.1

Related Lessons
INT0070509 OPS Tech. Spec. 3.8, Electrical Power Systems INT0239960 OPS SCR Events 2003

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.
INT02399600010600	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required. (INT0070509 EO 3)

Related References	
10CFR55.43	(B)(2)
TS	3.8.1

Related Skills (K/A)		ROI	SROI
2.2.22	Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2) (4.0/4.7) (/)		
2.2.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2) (3.2/4.2) (/)		
2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13) (3.1/4.2) (/)		

QUESTION: S 13

The plant is at 100% power with the following:

November 1 at 08:00: DG#2 is declared INOPERABLE for maintenance.
November 7 at 08:00: The ESST is declared INOP due to degraded voltage.
November 7 at 09:00: DG#2 is declared OPERABLE.
November 13 at 07:00: DG#1 is declared INOPERABLE due to low jacket water level.
November 13 at 08:00: The ESST is declared OPERABLE.

What date and time must you enter **the condition to be in** MODE 3 in 12 hours?

a. November 13 at 09:00.

b. November 14 at 08:00.

c. November 15 at 08:00.

d. November 20 at 07:00.

ANSWER: S 13

c. November 15 at 08:00.

TS 1.3, Completion Times. There is a separate completion time with a separate 14-day clock measuered from the time the LCO statement was declared not met at 08:00 on November 1. Fourteen (14) days later is 08:00 on November 15.

Distracter a. This is correct until the offsite circuit is declared operable at which point Condition D is exited and the clock for DG#1 inoperable for Condition B is continued (7-day clock) but there is a 14 day completion time for the LCO statement not met.
Distracter b. There is a separate completion time with a separate 14-day clock measuered from the time the LCO statement was declared not met at 08:00 on November 1. Fourteen (14) days later is 08:00 on November 15 not 14.
Distracter d. This is correct until the offsite circuit is declared operable at which point Condition D is exited and the clock for DG#1 inoperable for Condition B is continued (7-day clock) but there is a 14 day completion time for the LCO statement not met.

2004 Biennial Exam 04LORSRO03, Question # 21208

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	262001	G 2.2.7
	Importance Rating		3.6

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 0.26 SURVEILLANCE PROGRAM 61

Learning Objective: See Attached (As available)

Question Source:
- Bank # 20532
- Modified Bank # (Note changes or attach parent)
- New

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge L
- Comprehension or Analysis

10 CFR 55 Content
- 55.41
- 55.43 3

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 14 20532	02	01/26/2006	05/23/2010	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	N

Topic Area	Description
Administrative	INT0320101, Delegation of Authority to Authorize the Performance of Surveillances.

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

Related Objectives	
INT032010100G010I	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance test authorization

Related References	
PR	0.26

Related Skills (K/A)		ROI	SROI
2.2.7	Knowledge of the process for conducting special or infrequent tests. (CFR: 41.10 / 43.3 / 45.13) (2.9/3.6) (2.0 / 3.2)		

QUESTION: S 14

The station is required to perform a special test on 4160 VAC F Undervoltage relays due to a generic issue (Part 21) identified by the NRC.

While reviewing the Special Test Procedure you note the Shift Manager has signed that he gave the Management brief.

Does this meet the requirements for the management briefing required by Conduct of Infrequently Performed Tests or Evolutions, if not who must sign for this brief?

a. Yes; The Shift Manager can give and sign for this brief.

b. No; The Operations Manager must give and sign for this brief.

c. No; The Control Room Supervisor must give and sign for this brief.

d. No; The Shift Operations Supervisor must give and sign for this brief.

ANSWER: S 14

b. No; The Operations Manager must give and sign for this brief.

Explanation:

Procedure 2.0.1.1 Conduct of Infrequently Performed Tests or Evolutions states:

2.1 The Originator shall ensure the following items are addressed in the controlling document, as appropriate:

2 .1 .1 (Required) A sign-off for the General Manager of Plant Operations o r Operations Manager signifying completion of briefing on Management Expectations.

SRO Justification: 10CFR55.43 b (3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	215005 G 2.4.20	
	Importance Rating		4.3

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) EOP 7A
(Attach if not previously provided)
(including version/revision number) 14

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 15	00	1/08/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	N

Topic Area	Description
Emergency Operating Procedures	INT0080610, Actions after HSBW with power rise

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

Related Objectives	
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References	
CFR	10CFR55.43

Related Skills (K/A)		ROI	SROI
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:

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

- All Circ water pumps were lost.
- Very few control rods inserted on the scram.
- When the MSIVs closed, a small leak on a Recirc suction pipe occurred.
- Reactor pressure control was established using the SRVs.
- The crew initiated SLC.
- RPV water level was intentionally lowered.
- A level band of -183"(corrected FZ) to -60"(corrected FZ) was established.
- When Hot Shutdown Boron Weight had been injected, the crew began raising water level.

When the level reached -50"(corrected FZ) the following conditions were present:

• Reactor pressure	950 psig (steady)
• All APRMs are approximately	15% (rising)
• Drywell pressure	3.0 psig (slowly lowering)
• Torus pressure	3.5 psig (slowly lowering)
• Average Torus water temperature	160ºF (rising slowly)
• Average Drywell temperature	255ºF (slowly lowering)
• Primary Containment water level	12.8' (steady)

What action is required?

a. Emergency Depressurize the RPV IAW EOP 3A.

b. Reduce reactor pressure to less than 800 psig IAW EOP 6A.

c. Stop and prevent injection except for RCIC, CRD and SLC IAW EOP 7A.

d. Restore and maintain RPV water level between +3" and 54" IAW EOP 7A.

ANSWER:

c. Stop and prevent injection except for RCIC, CRD and SLC IAW EOP 7A.

Even though hot shutdown boron weight has been injected, reactor water level is greater than -60"(FZ) and reactor power is greater than 3%. EOP 7A has a note that if reactor power rises and continues to rise (as given) then they must reenter EOP 7A at "H". With reactor power above 3% they must transition to L and reevaluate. Therefore the crew is required to stop and prevent injection and lower water until level is less than -60"(FZ) AND one of the level power condition no longer exists.

Distractors:

a. no ED condition present.
b. Operation is in the safe region of the HCTL and CSBW has not been injected so cooldown is not allowed.
d. Power is continuing to rise therefore the note would require the crew to stop and prevent to lower level.

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		2
K/A #	223001.A2.07	
Importance Rating		4.3

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): EOP 3A, 13

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 16	00			NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	4	Multiple Choice	N

Topic Area	Description
Systems	COR0020302, Effect of High Drywell Pressure on Drywell Ventilation (2008 Requal EXAM 2006 NRC EXAM)

Related Lessons	
COR0020302	OPS CONTAINMENT

Related Objectives	
COR0020302001130D	Describe the PCIS design features and/or interlocks that provide for the following: Bypassing of selected isolations
COR0020302001130E	Describe the PCIS design features and/or interlocks that provide for the following: Operator action to defeat/reset isolations
COR0020302001170A	Predict the consequences of the following items on Primary containment: LOCA
COR0020302001210C	Given plant conditions, determine if the following should have occurred: Drywell cooling fan trip.

Related References	
10CFR55.43	(B)(5)
PR	5.8.10

Related Skills (K/A)		ROI	SROI
295024.EK2.18	Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: (CFR: 41.7 / 45.8) Ventilation. (3.3/3.4) (3.3 / 3.4)		
223001.A2.07	Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the conseq...: (CFR: 41.5 / 45 (High drywell pressure (4.2*/4.3* (4.2 / 4.3)		

QUESTION: S 16

A small break LOCA has occurred with the following conditions:

- Reactor water level is +45 inches (NR).
- Reactor pressure is 560 psig.
- Drywell pressure is 3.1 psig.
- Drywell temperature is 195☐F.
- Drywell FCU control switches are in RUN.

What is the status of DW cooling <u>and</u> what action is required?

a. All DW FCUs are running; vent primary containment IAW 2.4PC .

b. NO DW FCUs are running; Place all DW FCUs to Override IAW 2.4PC.

c. All DW FCUs are running; vent primary containment IAW EOP 3A.

d. NO DW FCUs are running; Place all DW FCUs to Override IAW EOP 3A.

ANSWER:

d. NO DW FCUs are running; Place all DW FCUs to Override IAW EOP 3A.

<u>Explanation:</u>

With a 1.84 psig signal in to trip the DW FCUs the control switches must be taken to override to clear that signal to get them to run. EOP 3A give the guidance to place DW FCU to override.

<u>Distractors:</u>

a. FCUs have tripped.

b. the drywell hi pressure signal can be overridden with the override switch when in the EOPs, but not in 2.4PC

c. FCUs have tripped.

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	268000	G 2.1.23
	Importance Rating		4.4

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 8.8.11 Liquid Radioactive Waste Discharge Authorization
(Attach if not previously provided)
(including version/revision number) 29

Learning Objective: See Attached (As available)

Question Source: Bank # 23510
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam 2006 CNS

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

10 CFR 55 Content 55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 17 23510	00	04/02/2004		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	N

Topic Area	Description
Administrative	Liquid Release Authorization/Approval and Actions for a Lost CW Pump During Release (ILT 2006 NRC EXAM)

Related Lessons	
INT0320115	OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115B0B0100	State who, by title, authorizes releases of radioactive liquid effluents from CNS.
INT0320115B0B0300	State the number of Circulating Water Pumps required to be in service during liquid radioactive discharges.

Related References	
10CFR55.43	(B)(4)
PR	8.8.11

Related Skills (K/A)		ROI	SROI
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: S 17

The plant is operating at low power with 2 Circulating Water pumps running. **De-icing** is in progress. The Radwaste Operator indicates that the Floor Drain Sample Tank requires discharging.

1) Whose approvals/authorizations is/are required in order to accomplish this discharge?
2) If one of the two operating circulating water pumps trip during the discharge, what action, if any, is required and why?

a. Chemistry department authorizes the release and the duty Shift Manager approves the release.
Continue the discharge sufficient dilution flow exists.

b. Duty Shift Manager authorizes and approves the release.
Continue the discharge sufficient dilution flow exists.

c. Chemistry department authorizes the release and the duty Shift Manager approves the release.
Terminate the discharge insufficient dilution flow exists.

d. Duty Shift Manager authorizes and approves the release.
Terminate the discharge insufficient dilution flow exists.

ANSWER: S 17

c. Chemistry department authorizes the release and the duty Shift Manager approves the release.
Terminate the discharge insufficient dilution flow exists.

Answer c is correct because procedure 8.8.11 requires that chemistry authorizes the release and the duty Shift Manager approves the release. The loss of one CW pump would reduce flow to less than the minimum required and the discharge should be terminated.

Distractors:

a. is incorrect because the discharge should be terminated.
b. is incorrect because the discharge should be terminated and the chemistry department authorizes the release.
d. is incorrect because chemistry authorizes the release.

SRO Justification: This is an SRO only item because in accordance with procedure 8.8.11 only the duty Shift Manager can approve liquid radioactive releases.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	219000.A2.16	
	Importance Rating		3.2

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) EOP 3A and EOP 6 A
(Attach if not previously provided)
(including version/revision number) 13 14

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41 ______
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 18	00			NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	3	Multiple Choice	N

Topic Area	Description
Emergency Operating Procedures	INT0080613, Actions High PC Water Level.

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.
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
PR	5.8

Related Skills (K/A)		ROI	SROI
219000.A2.16	Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal condition or operations: (CFR: 41.5/45.6) High suppression pool level (2.9/3.2)		

QUESTION: S 18

The plant was operating when a LOCA with an ATWS has occurred.

The following plant conditions are present:

- RPV water level is -60 inches wide range and rising.
- PCIS Group 1, 2, 3, 6, 7 have occurred.
- Reactor Pressure is 300 psig and slowly lowering.
- Drywell pressure is 20 psig and slowly lowering.
- Drywell Temperature is 220º F and slowly lowering.
- Suppression pool temperature is 160º F and slowly rising.
- Primary containment water level is 16.5 feet and slowly rising.
- HPCI has been stopped and prevented.
- Condensate and Feed has been stopped and prevented.

What action is required next?

a. Open 6 SRVs IAW EOP 6B.

b. Place all RHR and CS pumps in PTL IAW EOP 7A.

c. Maximize Suppression Pool cooling using all RHR pumps IAW EOP 3A.

d. Rapidly depressurize RPV with main turbine BPVs (disregard cooldown rate) IAW EOP 6A.

ANSWER: S 18

b. Place all RHR and CS pumps in PTL IAW EOP 6B.

Per EOP 3A Primary Containment Control when torus water level reaches 16.5 feet regardless of if it could be restored ED is required as this would violate the HTCL curve.

When in an ATWS EOPs require stop and prevent to be complete prior to opening 6 SRVs for ED.

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.4	
	Importance Rating		3.8

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) T.S Section 5.2
(Attach if not previously provided)
(including version/revision number) 200

Learning Objective: See Attached (As available)

Question Source: Bank # 24480
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41
55.43 2

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 19 24480	00	12/12/2008	12/20/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	2	Multiple Choice	N

Topic Area	Description
Technical Specifications, ODAM, TRM	T.S. 5.2.2 Unit Staff

Related Lessons
INT0070513 CNS Technical Specifications 5.0, Administrative Controls

Related Objectives	
INT00705130010100	Given a set of plant conditions, recognize non-compliance with a Chapter 5.0 Requirement.

Related References	
TS	5.2.2

Related Skills (K/A)		ROI	SROI
2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, ¿no-solo¿ operation, maintenance of active license status, 10CFR55, etc. (CFR: 41.10 / 43.2) (3.3/3.8) (2.3 / 3.4)		

QUESTION: S 19

The plant is preparing to return to power operations after a 30 day refueling outage. Minimum staffing requirements for Mode 5 in accordance with Technical Specifications has been set during the outage.

According to Technical Specifications, what additional personnel are required to operate in Mode 3?

a. One NLO, One SRO

b. One NLO, Two SROs, One RP

c. Two NLOs, One SRO, One STE

d. Two NLOs, One SRO, One RO, One STE

ANSWER: S 19

d. is correct - two additional NLO, an SRO,an RO, and an STE are required by T.S.

a. is incorrect - requires two additional NLOs and an STE
b. is incorrect - requires two additional NLOs and an STE. An RP was already assigned.
c. is incorrect - Need an RO.

SRO Justification: 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases.

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		2
K/A #	2.2.1	
Importance Rating		4.4

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) Procedure 2.1.1 and 10.13
(Attach if not previously provided)
(including version/revision number) 159 64

Learning Objective: See Attached (As available)

Question Source:

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

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	L
Comprehension or Analysis	

10 CFR 55 Content

55.41	
55.43	5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 20	00	1/8/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	2	Multiple Choice	N

Topic Area	Description
Integrated Plant	Period precautions and limits approach to crit.

Related Lessons
SKL0124301 OPS COLD STARTUP (All Rods In to 100% Power)

Related Objectives	
SKL01243010010200	Given a simulated Reactor plant with a startup in progress, the crew will withdraw Control Rods in accordance with procedure 10.13 to bring the Reactor critical and calculate the Reactor period in accordance with procedure 2.1.1.

Related References
2.1.1 STARTUP PROCEDURE 10.13 CONTROL ROD SEQUENCE AND MOVEMENT CONTROL

Related Skills (K/A)		ROI	SROI
2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13) (4.4/4.7)		
2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1) (4.5/4.4)		

QUESTION:

Given the following set of conditions:

- A Reactor startup in progress.
- The RO withdraws control rod 22-19 from notch 12 to notch 14.
- Reactor period changes from 100 seconds to a stable 39 seconds.

Which of the following identifies the NEXT required action to be taken?

a. Re-insert control rods as necessary to achieve sub-criticality per GOP 2.1.1 Startup Procedure.

b. Re-insert control rods as necessary to achieve sub-criticality per NPP 10.13 Control Rod Sequence.

c. Re-insert control rod 22-19 to obtain a stable period indication of greater than 50 seconds per GOP 2.1.1 Startup Procedure.

d. Re-insert control rod 22-19 to obtain a stable period indication of greater than 50 seconds per NPP 10.13 Control Rod Sequence.

ANSWER:

c. Re-insert control rod 22-19 to obtain a stable period indication of greater than 50 seconds per GOP 2.1.1 Startup Procedure.

Explanation: Per GOP 2.1.1 "A period faster than 50 seconds shall not be maintained." The operator would reinsert 22-19 to notch 12 to establish stable period that is longer than 50 seconds. There is no requirement to continue inserting control rods to achieve a sub-critical condition.

REQUIRED REFERENCES: None

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:

Level	RO		SRO
Tier #			3
Group #			3
K/A #		2.3.13	
Importance Rating			3.8

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): 9.EN-RP-108

5

Learning Objective: See Attached (As available)

Question Source:

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

Question History: Last NRC Exam NA

Question Cognitive Level:

Memory or Fundamental Knowledge	L
Comprehension or Analysis	

10 CFR 55 Content

55.41	
55.43	5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 21	00	01/08/2011		Licensed Operator	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	N

Topic Area	Description
Administrative	INT0320115, (DDE) of 1250 mrem/hour. How should the entrance to this room be posted?

Related Lessons
INT0320115 OPS CNS Administrative Procedures Radiation Protection and Chemistry Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT0320115H0H010D	Discuss the following as described in Radiological Protection Procedure 9.EN-RP-108, Area Posting and Access Control: RCA/Satellite Area controls and postings

Related References	
PR	9.EN-RP-108
TS	5.7

Related Skills (K/A)		ROI	SROI
2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10) (3.2/3.7) (2.5 / 3.1)		
2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, (CFR: 41.12 / 43.4 / 45.9 / 45.10) (3.4/3.8)		

QUESTION: S 21

One of the rooms in the Reactor Building contains an area in which a person could receive a deep dose equivalent (DDE) of 1250 mRem/hour.

How should the entrance to this room be posted?

a. CAUTION - HIGH RADIATION AREA IAW 9.EN-RP-108 Radiation Protection Posting

b. CAUTION - HIGH RADIATION AREA IAW CNS Technical Specification 5.7.1 High Radiation Area.

c. CAUTION - LOCKED HIGH RADIATION AREA IAW 9.EN-RP-108 Radiation Protection Posting

d. CAUTION - LOCKED HIGH RADIATION AREA IAW Technical Specification 5.7.1 High Radiation Area.

ANSWER: S 21

c. CAUTION - LOCKED HIGH RADIATION AREA IAW 9.EN-RP-108 Radiation Protection Posting

9.EN-RP-108 Radiation Protection Posting, Locked High Radiation Area (LHRA) - An area, accessible to individuals, in which radiation levels from sources external to the body could result in an individual receiving a deep dose equivalent >_ 1 Rem (10 mSv) in 1 hour at 30 cm (- 12") from the radiation source or from any surface that the radiation penetrates.

This is not covered in 5.7.1 which is each high radiation area in which the deep dose equivalent in excess of 100 mrem but less than 1000 mrem in one hour (measurement made at 12 inches from source of radiation) shall be barricaded (barricade will impede physical movement across the entrance or access to the high radiation area; i.e., doors, yellow and magenta rope, turnstile) and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP).
Per Technical Specifications, section 5.7.2, an area in which an individual could pick up a DDE in excess of 1000 mem in 1 hour shall be classified as a locked high radiation area.

Reference: Procedure 9.EN-RP-108, Technical Specifications

SRO Justification: 10CFR55.43 b (2) Facility operating limitations in the technical specifications and their bases.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.28	
	Importance Rating		4.1

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 5.7.1 Attachment 1 EAL Matrix 43

Learning Objective: See Attached (As available)

Question Source:
- Bank # 20682
- Modified Bank # (Note changes or attach parent)
- New

Question History: Last NRC Exam NA

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis H

10 CFR 55 Content
- 55.41
- 55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 22 20682	00	04/21/2004	05/23/2010	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	H	1	5	Multiple Choice	N

Topic Area	Description
Emergency Plan	INT0320136, SECURITY EAL

Related Lessons
COR0090001 OPS ACP/EP REVIEW INT0320136 OPS-CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
COR00900010010400	Given plant conditions and appropriate Abnormal/Emergency Procedure, determine required Subsequent Operator Actions(s).

Related References	
PR	5.7.1

Related Skills (K/A)		ROI	SROI
2.4.28	Knowledge of procedures relating to a security event (non-safeguards information). (CFR: 41.10 / 43.5 / 45.13) (3.2/4.1) (2.3 / 3.3)		

QUESTION: S 22

The plant was operating at near rated power when the Security Shift Supervisor informed the Shift Manager that heavily armed intruders were attempting to gain entry into the protected area. The reactor was scrammed and the Station Operators directed to obtain fire fighting gear and report to the Control Room. The following events occurred following initial report by the Security Shift Supervisor:

- 0900 Armed intruders have overpowered all the Security Force that is oustide of CAS and SAS.
- 0905 An explosion occurs in #1 Diesel Generator Room approximately 1 minute after the intruders gain entry into that room.
- 0910 Intruders are entering the Reactor Building 903 southeast via the steam tunnel roof and the ASD room door.
- 0915 Intruders occupy CAS, SAS and have taken the entire operating crew hostage.

When is a General Emergency first required to be declared?

a. 0900

b. 0905

c. 0910

d. 0915

ANSWER: S 22

c. 0910

The intruders have gained access to the reactor building via ASD room which they must now control. This requires the declaration of a General Emergency. The fact that the intruders occupy ASD would indicated that station control is lost and a General Emergency is required.

Distractors:

a. is incorrect as this would only require the declaration of an alert.

b. is incorrect as this would only require the declaration of a SAE.

d. is incorrect although this requires a GE this is not the first condition requiring a GE as asked in the test item stem.

PROVIDE THE STUDENTS Copy of Procedure 5.7.1 Attachment 1

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.43	
	Importance Rating		3.3

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) (Attach if not previously provided) (including version/revision number): Procedure 2.0.3 and 2.3.1
72 58

Learning Objective: See Attached (As available)

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

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 23	00	NEW		NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	4	Multiple Choice	N

Topic Area	Description
Administrative	INT0320103, Annunciator Procedure

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300E010D	Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Annunciator disabling

Related References	
PR	2.3.1

Related Skills (K/A)		ROI	SROI
2.2.43	Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13) (3.0/3.3)	3.0	3.3

QUESTION: S 23

The plant is operating at near rated power when the following annunciator alarms:

A-2/C-2 RFP TURBINE B LOW VACUUM PRE-TRIP

- The crew reviews plant conditions and determines the alarm is invalid, but will not reset.
- Further investigation by maintenance reveals the parts to fix the alarm will not be on site for 5 days.
- The decision is made to disable the alarm.

What is the acceptable method for tracking this disabled alarm?

a. Tracked on the Crew turnover sheet IAW COP 2.0.3 Conduct of Operations.

b. Tracked in the NOMs Narrative logs as a night order IAW Operations Instruction #4 Standing and Night Orders.

c. Tracked in the NOMs LCO tracking module IAW AP 2.3.1 General Alarm Procedure.

d. Tracked in the NOMs Narrative logs as a narrative log entry IAW COP 2.0.2 Operations Logs and Reports.

ANSWER: S 23

c. Tracked in the NOMs LCO tracking module IAW AP 2.3.1 General Alarm Procedure. .

Explanation- Per Procedure 2.3.1 General Alarm Procedure step 7.4 Disabled alarm evaluation shall be documented and tracked by NOMS LCO Tracking module per Operations Desk Guide 3, NOMS, or Attachment 1, Alarm Problem Evaluation (APE), if NOMS is not available. Alarms disabled for short durations (24 hours) maybe tracked with a narrative log entry. The alarm evaluation and compensatory actions should be specified in the log entry.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.32	
	Importance Rating		4.0

Proposed Question: See Attached

Proposed Answer: See Attached

Explanation: See Attached

Technical Reference(s) 2.2.9
(Attach if not previously provided)
(including version/revision number) 71

Learning Objective: See Attached (As available)

Question Source: Bank # 24571
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam NA

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

10 CFR 55 Content 55.41 ______
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 24 24571	00	02/12/2009	05/23/2010	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	L	1	3	Multiple Choice	N

Topic Area	Description
Systems	COR0020602, Stroke requirements following manual operation of a CS valve.(2008 Requal Exam)

Related Lessons	
COR0030301	Valves (GP)

Related Objectives	
COR00303010001500	Describe the precautions used while manually operating a motor-operated valve.

Related References	
10CFR55.41	(B)(7)
PR	2.2.9

Related Skills (K/A)		ROI	SROI
209001.A4.03	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Injection valves (3.7 / 3.6)		
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.8/4.0) (/)		

QUESTION: S 24

The plant is operating at rated power with the following conditions:

- CS Inboard Injection Valve (MO-12A) is closed.
- CS Outboard Injection Valve (MO-11A) is de-energized and is open (valve was opened using the manual operator).

After CS-MO-11A is re-energized what actions are required?

a. Locally using the manual operator fully close then reopen CS-MO-11A IAW SOP 2.2.9 Core Spray System.

b. From the control room fully close then reopen CS-MO-11A IAW SOP 2.2.9 Core Spray System.

c. Locally using the manual operator fully close then reopen CS-MO-11A IAW 6.1CS.203 CS-MO-12A Operability Test with Reactor Pressure >450 PSIG (IST) (DIV 1)

d. From the control room fully close then reopen CS-MO-11A IAW 6.1CS.203 CS-MO-12A Operability Test with Reactor Pressure >450 PSIG (IST) (DIV 1)

ANSWER: S 24

b. From the control room fully close then reopen CS-MO-11A IAW SOP 2.2.9 Core Spray System.

Explanation:

Procedure 2.2.9 caution requires that following manual engagement and operation of any essential limitorque motor operated valve, it shall be operated electrically to ensure electrical gear train is engaged. This operation may be a partial stroke, but CS-MO-11A is a seal in Valve. Since MO-12A is already closed and in its standby status position only MO-11A need only be moved electrically and then left in its standby position (open).

Distractors:

a. is incorrect because electrically stroking the valve is required.

c. is incorrect because electrically stroking the valve is required and this is not the correct reference procedure.

d. this is not the correct reference procedure.

SRO Justification: 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.5	
	Importance Rating		4.3

Proposed Question: See Attached

Proposed Answer: _See Attached_

Explanation: See Attached

Technical Reference(s) Admin Procedure 0.1 Procedure Use And Adherence
(Attach if not previously provided)
(including version/revision number) 35

Learning Objective: See Attached (As available)

Question Source:
Bank # 23314
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam NA

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

10 CFR 55 Content
55.41
55.43 5

Comments:

Question Number	Revision Number	Revision Date	Last Used Date	Exam Bank	Applicability	
S 25 23314	01	08/06/2009	01/07/2011	Licensed Operator	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	L	1	3	Multiple Choice	N

Topic Area	Description
Administrative	INT0320101, Explain how to perform actions when an abnormal conflicts with an EOP. (2006 ILT AUDIT EXAM)

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

Related Objectives	
INT032010100E010A	Discuss the following as described in Administrative Procedure 0.1, Introduction to CNS Operations Manual: Procedural adherence
INT032010100E010E	Discuss the following as described in Administrative Procedure 0.1, Introduction to CNS Operations Manual: Procedure hierarchy

Related References	
PR	0.1

Related Skills (K/A)		ROI	SROI
2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (CFR: 41.10 / 43.5 / 45.13) (3.7/4.3) (2.9 / 3.6)		

QUESTION: S 25

One of the Immediate Operator Actions of 2.4VAC as condenser vacuum degrades is to close the MSIVs if vacuum cannot be maintained $\geq$ 12" Hg. Due to a loss of alternate pressure control systems the Bypass Valves are the only system available to perform EOP 1A Pressure Control actions.

How is the Senior Reactor Operator to respond to this conflict between the procedures?

a. Invoke 10CFR50.54X and perform the actions to reopen the MSIVs even though the Abnormal directs that they be closed.

b. Get authorization from two SROs to deviate from the Abnormal and reopen the MSIVs to perform the actions in EOP 1A.

c. Follow the actions of EOP 1A and open the MSIVs, disregard the actions in AP 2.4VAC that require them to be closed as vacuum lowers.

d. Contact the work control center to expedite repairs of the Alternate Pressure Control systems and wait until they are available to continue EOP 1A actions.

ANSWER: S 25

c. Follow the actions of EOP 1A and open the MSIVs, disregard the actions in AP 2.4VAC that require them to be closed as vacuum lowers.

Explanation:

Ops Policy Procedure 2.0.1.2

Alarm/Abnormal/Emergency/System Operating Procedures/Instrument Operating Procedures may be carried out concurrently with an EOP. In the event that conflicting actions are directed by procedures, the EOP actions shall take precedence. EOPs/SAMGs are Operations highest tier procedure. If an explicit operation is directed by EOPs per a 5.8 EOP Support Procedure, then transition shall be made from the Alarm/Abnormal/Emergency/System Operating/Instrument Operating Procedures (including hard cards) to the 5.8 Procedure to perform or continue performing that operation.

SRO Justification 10CFR55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Facility: Cooper Nuclear Station		Date of Exam: 2-28-2011														
Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	4	3	3	N/A			3	4	N/A		3	20	3	4	7
	2	1	1	1				1	1			2	7	2	1	3
	Tier Totals	5	4	4				4	5			5	27	6	4	10
2. Plant Systems	1	2	2	2	3	2	2	2	3	3	3	2	26	2	3	5
	2	1	1	2	1	1	1	1	1	1	1	1	12	1 1	1	3
	Tier Totals	3	3	4	4	3	3	3	4	4	4	3	38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4			1 2	3 4	
					3		2		2		3		10	2 2	1 2	7

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics’ importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 RO

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.05 Ability to determine and/or interpret Jet pump operability as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:	3.1	53
295003 Partial or Complete Loss of AC / 6			X				AK3.06 Knowledge of the reasons for Containment isolation as they apply to PARTIAL OR COMPLETE LOSS OF A.C.POWER	3.7	46
~~295004 Partial or Total Loss of DC Pwr / 6~~						~~X~~	~~2.1.27 Knowledge of system purpose and/or function.~~	~~3.9~~	~~56~~
295004 Partial or Total Loss of DC Pwr / 6						X	2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).	3.4	56
295005 Main Turbine Generator Trip / 3		X					AK2.01 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and RPS	3.8	43
295006 SCRAM / 1						X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	54
295016 Control Room Abandonment / 7				X			AA1.02 Ability to operate and/or monitor the reactor/turbine pressure regulating system as they apply to CONTROL ROOM ABANDONMENT:	2.9	49
295018 Partial or Total Loss of CCW / 8		X					AK2.01 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and system loads	3.3	42
295019 Partial or Total Loss of Inst. Air / 8		X					AK2.01 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and CRD hydraulics	3.8	44
~~295021 Loss of Shutdown Cooling / 4~~			~~X~~				~~AK3.03 Knowledge of the reasons for increasing drywell cooling as they apply to LOSS OF SHUTDOWN COOLING~~	~~2.9~~	~~45~~
295021 Loss of Shutdown Cooling / 4			X				AK3.01 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Raising reactor water level	3.3	45
295023 Refueling Acc / 8	X						AK1.03 Knowledge of the operational implications of inadvertent criticality as they apply to REFUELING ACCIDENTS	3.7	41
295024 High Drywell Pressure / 5	X						EK1.01 Knowledge of the operational implications of drywell integrity as they apply to HIGH DRYWELL PRESSURE	4.1	40
~~295025 High Reactor Pressure / 3~~					~~X~~		~~EA2.03 Ability to determine and/or interpret Suppression pool temperature as they apply to HIGH REACTOR PRESSURE:~~	~~3.9~~	~~57~~
295025 High Reactor Pressure / 3					X		EA2.02 Ability to determine and/or interpret Reactor power as they apply to HIGH REACTOR PRESSURE:	4.2	57
295026 Suppression Pool High Water Temp. / 5					X		EA2.02 Ability to determine and/or interpret Suppression pool level as they apply to SUPPRESSION POOL HI WTR TEMPERATURE:	3.8	51
~~295027 High Containment Temperature / 5~~									
295028 High Drywell Temperature / 5	X						EK1.01 Knowledge of the operational implications of the reactor water level measurement as they apply to HIGH DRYWELL TEMPERATURE	3.5	39
295030 Low Suppression Pool Wtr Lvl / 5				X			EA1.05 Ability to operate and/or monitor HPCI as they apply to LOW SUPPRESSION POOL WATER LEVEL:	3.5	50
~~295031 Reactor Low Water Level / 2~~	~~X~~						~~EK1.03 Knowledge of the operational implications of water level effects on reactor power as they apply to REACTOR LOW WATER LEVEL:~~	~~3.7~~	~~58~~
295031 Reactor Low Water Level / 2	X						EK1.01 Knowledge of the operational implications of Adequate Core Cooling as they apply to REACTOR LOW WATER LEVEL	4.6	58
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X				EK3.03 Knowledge of the reasons for lowering reactor water level as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:	4.1	47

~~295038 High Off-site Release Rate / 9~~						~~X~~	~~2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.~~	~~3.1~~	~~55~~
295038 High Off-site Release Rate / 9						X	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.	4.2	55
600000 Plant Fire On Site / 8					X		AA2.05 Ability to determine and interpret Ventilation alignment necessary to secure affected area as they apply to PLANT FIRE ON SITE:	2.9	52
700000 Generator Voltage and Electric Grid Disturbances / 6				X			AA1.01 Ability to operate and/or monitor Grid frequency and voltage as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES:	3.6	48
K/A Category Totals:	4	3	3	3	4	3	Group Point Total:		20

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 RO

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3		X					AK2.08 Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the Condenser circulating water system	3.1	60
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2	X						295008.AK1.03 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: Main turbine trip	3.4	59
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1			X				AK3.01 Knowledge of the reasons for bypassing rod insertion blocks as they apply to INCOMPLETE SCRAM	3.4	61
295017 High Off-site Release Rate / 9									
~~295020 Inadvertent Cont. Isolation / 5 & 7~~	~~X~~						~~AK1.04 Knowledge of the operational implications of Bottom head thermal stratification as they apply to INADVERTENT CONTAINMENT ISOLATION:~~	~~2.5~~	~~59~~
295022 Loss of CRD Pumps / 1					X		AA2.01 Ability to determine and/or interpret accumulator pressure as they apply to LOSS OF CRD PUMPS:	3.5	63
295029 High Suppression Pool Wtr Lvl / 5						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps.	4.6	65
295032 High Secondary Containment Area Temperature / 5				X			EA1.05 Ability to operate and/or monitor affected systems so as to isolate damaged portions as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE:	3.7	62
295033 High Secondary Containment Area Radiation Levels / 9						X	2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	3.8	64
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:	1	1	1	1	1	2	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 1 RO

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode					X							K5.02 Knowledge of the operational implications of core cooling methods as they apply to RHR/LPCI INJECTION MODE.	3.5	9
205000 Shutdown Cooling						X						K6.04 Knowledge of the effect that a loss or malfunction of reactor water level will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE):	3.6	11
206000 HPCI							X					A1.08 Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including system lineup	4.1	14
~~207000 Isolation (Emergency) Condenser~~														
~~207000 Isolation (Emergency) Condenser~~														
209001 LPCS				X								K4.01 Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for prevention of over pressurization of core spray piping	3.2	7
~~209002 HPCS~~														
211000 SLC										X		A4.07 Ability to manually operate and/or monitor in the control room: Lights and alarms	3.6	23
212000 RPS								X				A2.08 Ability to (a) predict the impacts of Low reactor level on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	4.1	15
215003 IRM					X							K5.01 Knowledge of the operational implications of detector operation as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM:	2.6	10
215004 Source Range Monitor		X										K2.01 Knowledge of electrical power supplies to the SRM channels/detectors	2.6	4
215005 APRM / LPRM			X									K3.07 Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on the rod block monitor:	3.2	5
215005 APRM / LPRM									X			A3.02 Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Full core display	3.5	17
217000 RCIC			X									K3.02 Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on Reactor vessel pressure	3.6	6
217000 RCIC									X			A3.04 Ability to monitor automatic operations of the REACTOR CORE ISOL COOLING SYSTEM (RCIC) including, system flow.	3.6	18
218000 ADS											X	2.1.19 Ability to use plant computers to evaluate system or component status.	3.9	22
223002 PCIS/Nuclear Steam Supply Shutoff										X		A4.02 Ability to manually operate and/or monitor in the control room: Manually initiate the system	3.9	19
~~239002 SRVs~~											~~X~~	~~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.~~	~~4.0~~	~~21~~

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 1 RO

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
239002 SRVs											X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	21
239002 SRVs								X				A2.03 Ability to (a) predict the impacts of Stuck open SRV on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	4.1	26
259002 Reactor Water Level Control										X		A4.07 Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from automatic to manual mode	3.8	20
261000 SGTS	X											K1.12 Knowledge of the physical connections and/or cause/effect relationships between STANDBY GAS TREATMENT SYSTEM and the primary containment purge system:	3.1	2
261000 SGTS								X				A2.04 Ability to (a) predict the impacts of High train moisture content on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	2.5	16
262001 AC Electrical Distribution		X										K2.01 Knowledge of electrical power supplies to the following: Off-site sources of power	3.3	3
262001 AC Electrical Distribution				X								K4.03 Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the interlocks between automatic bus transfer and breakers	3.1	25
262002 UPS (AC/DC)				X								K4.01 Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for transfer from preferred power to alternate power supplies.	3.1	8
263000 DC Electrical Distribution	X											K1.03 Knowledge of the physical connections and/or cause/effect relationships between D.C. ELECTRICAL DISTRIBUTION and battery ventilation:	2.6	1
264000 EDGs							X					A1.04 Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including crank case temperature and pressure	2.6	13
~~300000 Instrument Air~~									~~X~~			~~A3.02 Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including air temperature~~	~~2.9~~	~~17~~
400000 Component Cooling Water						X						K6. 07 Knowledge of the effect that a loss or malfunction of Breakers, relays, and disconnects will have on the CCWS	2.7	12
400000 Component Cooling Water									X			A3.01 Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	3.0	24
~~400000 Component Cooling Water~~								~~X~~				~~A2.04~~	~~2.9~~	~~16~~
K/A Category Point Totals:	2	2	2	3	2	2	2	3	3	3	2	Group Point Total:		26

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 2 RO

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS														
201003 Control Rod and Drive Mechanism				X								K4.08 Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: Monitoring CRD mechanism temperature	2.6	30
~~201004 RSCS~~														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control		X										K2.02 Knowledge of electrical power supplies to the following: Hydraulic power unit:	2.6	28
204000 RWCU														
214000 RPIS						X						K6.01 Knowledge of the effect that a loss or malfunction of the A.C. electrical power will have on the ROD POSITION INFORMATION SYSTEM:	2.5	32
215001 Traversing In-core Probe								X				A2.07 Ability to (a) predict the impacts of failure to retract during accident conditions on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations	3.4	34
215002 RBM														
216000 Nuclear Boiler Inst.			X									K3.29 Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on Jet pump flow monitoring:	3.1	38
219000 RHR/LPCI: Torus/Pool Cooling Mode			X									K3.01 Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS / SUPPRESSION POOL COOLING MODE will have on Suppression pool temperature control	3.9	29
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode	X											K1.01 Knowledge of the physical connections and/or cause/effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE and the following: Suppression pool	3.6	27
233000 Fuel Pool Cooling/Cleanup					X							K5.07 Knowledge of the operational implications of the Maximum (abnormal) heat 102d load as they apply to FUEL POOL COOLING AND CLEAN-UP:	2.5	31
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator							X					A1.06 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: Main turbine steam flow 3.2	3.2	33
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate									X			A3.05 Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: Lights and alarms	3.0	35

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 2 RO

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
259001 Reactor Feedwater														
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
~~290001 Secondary CTMT~~										~~X~~		~~A4.04 Ability to manually operate and/or monitor in the control room: Auxiliary building area temperature:~~	~~2.6~~	~~36~~
290001 Secondary CTMT										X		A4.01 Ability to manually operate and/or monitor in the control room: Reactor building differential pressure: Plant-Specific	3.3	36
290003 Control Room HVAC											X	2.2.22 Knowledge of limiting conditions for operations and safety limits. (Note 1) ES-401-4 form	4.0	37
290002 Reactor Vessel Internals														
K/A Category Point Totals:	1	1	2	1	1	1	1	1	1	1	1	Group Point Total:		12

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 SRO		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 Partial or Complete Loss of AC / 6									
295004 Partial or Total Loss of DC Pwr / 6					X		AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of D.C. power.	3.6	76
295005 Main Turbine Generator Trip / 3									
295006 SCRAM / 1									
295016 Control Room Abandonment / 7									
295018 Partial or Total Loss of CCW / 8							AA2.03 Ability to determine and/or interpret the Cause for partial or complete loss as it apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER.	3.5	78
295019 Partial or Total Loss of Inst. Air / 8									
295021 Loss of Shutdown Cooling / 4									
295023 Refueling Acc / 8					X		AA2.05 Ability to determine and/or interpret the entry conditions of emergency plan as they apply to REFUELING ACCIDENTS.	4.6	77
295024 High Drywell Pressure / 5						X	2.4.6 Knowledge of EOP mitigation strategies.	4.7	80
~~295025 High Reactor Pressure / 3~~					~~X~~		~~EA2.03 Ability to determine and/or interpret suppression pool temperature as they apply to HIGH REACTOR PRESSURE:~~	~~4.1~~	~~78~~
~~295026 Suppression Pool High Water Temp. / 5~~					~~X~~		~~EA2.03 Ability to determine and/or interpret the Reactor Pressure as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE~~	~~4.0~~	~~76~~
~~295027 High Containment Temperature / 5~~						~~X~~	~~2.1.27 Knowledge of system purpose and/or function~~	~~4.0~~	~~81~~
295027 High Containment Temperature / 5						X	2.1.39 Knowledge of conservative decision making practices.	4.3	81
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2						X	2.4.3 Ability to identify post-accident instrumentation.	3.9	82
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9						X	2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	4.6	79
600000 Plant Fire On Site / 8									
700000 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:					3	4	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 SRO

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7						X	2.2.22 Knowledge of limiting conditions for operations and safety limits.	4.7	84
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5					X		EA2.01 Ability to determine and/or interpret Suppression pool water level as they apply to HIGH SUPPRESSION POOL WATER LEVEL. 3.9	3.9	85
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9					X		EA2.01 Ability to determine and/or interpret Area radiation levels as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS.	3.9	83
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:					2	1			3

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems - Tier 2/Group 1 SRO

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode														
205000 Shutdown Cooling								X				A2.09 Ability to (a) predict the impacts of Reactor low water level on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	3.8	86
206000 HPCI														
~~207000 Isolation (Emergency) Condenser~~														
209001 LPCS														
~~209002 HPCS~~														
211000 SLC														
212000 RPS								X				A2.19 Ability to (a) predict the impacts of Partial system activation (half-SCRAM) on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations.	3.9	87
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM											X	2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	4.3	90
217000 RCIC														
218000 ADS														
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs														
259002 Reactor Water Level Control														
261000 SGTS														
262001 AC Electrical Distribution											X	2.2.7 Knowledge of the process for conducting special or infrequent tests.	3.6	89
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs											X	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	88
K/A Category Point Totals:								2			3	Group Point Total:		5

ES-401 | BWR Examination Outline | Form ES-401-1

Plant Systems - Tier 2/Group 2 SRO

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS														
201003 Control Rod and Drive Mechanism														
~~201004 RSCS~~														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS														
215001 Traversing In-core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode								X				A2.16 Ability to (a) predict the impacts of high suppression pool level on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	3.2	93
223001 Primary CTMT and Aux.								X				A2.07 Ability to (a) predict the impacts of High drywell pressure on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	4.3	91
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														
259001 Reactor Feedwater														
268000 Radwaste											X	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. 4.4	4.4	92
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
K/A Category Point Totals:								2			1	Group Point Total:		3

Facility: Cooper Nuclear Station Date of Exam: 2-28-2011

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.5	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	2.9	66		
	2.1.17	2.1.17 Ability to make accurate, clear, and concise verbal reports.	3.9	67		
	2.1.27	2.1.27 Knowledge of system purpose and/or function.	3.9	75		
	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.			3.8	94
	2.1.32	Ability to explain and apply system limits and precautions.			4.0	99
	Subtotal			3		2
2. Equipment Control	2.2.14	2.2.14 Knowledge of the process for controlling equipment configuration or status.	3.9	69		
	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.			4.4	95
	2.2.43	Knowledge of the process used to track inoperable alarms.			3.3	98
	2.2.36	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1	68		
	~~2.2.4~~	~~(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.~~	~~3.6~~	~~68~~		
	Subtotal			1		2
3. Radiation Control	2.3.11	2.3.11 Ability to control radiation releases.	3.8	70		
	2.3.4	2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71		
	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.			3.8	96
	Subtotal			2		1
4. Emergency Procedures / Plan	2.4.43	2.4.43 Knowledge of emergency communications systems and techniques.	3.2	72		
	2.4.22	2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.	3.6	73		
	2.4.32	2.4.32 Knowledge of operator response to loss of all annunciators.	3.6	74		
	2.4.28	Knowledge of procedures relating to a security event (non-safeguards information).			4.1	97
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.			4.3	100
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1	207000.A3.02	Cooper does not have an isolation condenser
2/1	207000.K3.02	Cooper does not have an isolation condenser
2/1 SRO	207000	Cooper does not have an isolation condenser
2/1	209002.A2.08	Cooper does not have a high pressure core spray system
2/1 SRO	209002	Cooper does not have a high pressure core spray system
2/1	2.2.3	Cooper is not a multi-unit facility
2/2	201004.A2.01	Cooper no longer uses a Rod Sequence Control System
2/2 SRO	201004	Cooper no longer uses a Rod Sequence Control System
1/1	295027.EK1.02	Cooper has a Mark I containment design not a Mark III.
1/1	295038 Generic 2.2.36	This Generic K/A deals with the effects of maintenance on the status of LCOs – A psychometrically valid question could not be developed based on 10CFR55.41.
3	2.2.4	Cooper is not a multi-unit facility
2/2	2.4.45	Replaced Generic K/A with a randomly selected A2 K/A to fill all blocks on form 401-1. **Note 1**
2/1	400000.A2.04	Could not come up with a psychometrically valid question for this K/A, Also CCW was sampled three times in this group. Randomly selected another system.
1/1	295004 G 2.1.27	Unable to develop a psychometrically valid question for this K/A. Randomly selected another generic K/A.
1/1	295021.AK3.03	Unable to develop a psychometrically valid question for this K/A. Cooper has no actions for ventilation on a loss of SDC. Randomly selected another K/A.
1/1	295025.EA2.03	Unable to develop a psychometrically valid question for this K/A. Randomly selected another K/A.
1/1	295031.EK1.03	Unable to develop a psychometrically valid question for this K/A. This question and question 47 were too similar.
1/2	295020.AK1.04	Unable to develop a psychometrically valid question for this K/A. Randomly selected another system.
2/1	239002. G 2.4.21	Unable to develop a psychometrically valid question for this K/A. Randomly selected another K/A.
2/1	300000.A3.02	Unable to develop a psychometrically valid question for this K/A. Randomly selected another system.
2/2	290001.A4.04	Unable to develop a psychometrically valid question for this K/A. Cooper does not have any buildings named or designated as Auxiliary Building. Randomly selected another K/A.
1/1 SRO	295025.EA2.03	Unable to develop a psychometrically valid question for this K/A. Too similar to another RO question. Randomly selected another system.

1/1 SRO	295026.EA2.03	Unable to develop a psychometrically valid question for this K/A. Too similar to another RO question. Randomly selected another system.
1/1 SRO	295027. G 2.1.27	Unable to develop a psychometrically valid question for this K/A. Randomly selected another generic K/A.