Answer Key

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date: June 13, 2003	Facility/Unit: Cooper Nuclear Station
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time: 08:00	Finish Time: 14:00

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected six hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	___________	Points
Applicant's Score	___________	Points
Applicant's Grade	___________	Percent

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	1111	02	03/28/2003	06/13/2003	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	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001130J	Predict the consequences of the following events on the AC Electrical Distribution System: Exceeding current limitations

Related References	
2.2.18	4160V Auxiliary Power Distribution System
2.2.20	Procedure 2.2.20, Standby AC Power System (Diesel Generator)

Related Skills (K/A)	
262001.A4.05	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Voltage, current, power, and frequency on A.C. buses (3.3/3.3)

QUESTION: 1 1111 (point(s))

Given the following conditions:

- ➢ Reactor is in Hot Shutdown
- ➢ DG 1 is paralleled to 1F for surveillance testing
- ➢ The startup transformer supply breaker 1AS trips
- ➢ DG 1 load reaches 150% of rated current

Which breaker(s) will trip?

a. **ONLY** EG1

b. **ONLY** 1AF

c. **BOTH** 1AF and 1FA

d. **BOTH** EG1 and 1FA

ANSWER: 1 1111

c. **BOTH** 1AF and 1FA

1FA is tripped by the over current condition 1AF trips because 1AF is in NORMAL AFTER CLOSE and Bus 1A is deenergized.

Answer source: 2.2.18, pp. 170, 171 (1AF), step 2.6.2, p. 173 (1FA), step 2.9.2

Distractors:

a. EG1 does not trip.
b. Both 1FA and 1AF will be tripped.
d. EG1 Remains closed to maintain 1F energized.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	14036	00	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001070C	State the electrical power supplies to the following: PMIS Computer
COR0010102001080E	Predict the consequences of the following on plant operation: PMIS/UPS inverter failure
COR0010102001090C	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Automatic bus transfer
COR0010102001060D	Describe the interrelationship between the AC Electrical Distribution System and the following: PMIS/UPS

Related References	
2.2.63	Procedure 2.2.63, PMIS Uninterruptible Power Supply System

Related Skills (K/A)	
262002.K1.06	Knowledge of the physical connections and/or cause- effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Unit computer: Plant-Specific (2.6/2.7)

QUESTION: 2 14036 (1 point(s))

The plant was operating at power with the emergency transformer out of service. A fault occurred that resulted in the lockout of 4160V bus 1A and 4160V bus 1B. Both Diesel generators started and loaded their respective buses.

If MDP-2 were to deenergize at this time, what power, if any, would be immediately supplied to the PMIS computer?

PMIS would be . . .

a. deenergized.

b. powered directly from MDP-1.

c. powered from the 125 VDC PMIS battery via the inverter.

d. powered from MCC-L via the inverter and battery charger.

ANSWER: 2 14036

c. powered from the 125 VDC PMIS battery via the inverter.

The loss of the lockout experienced on the plant's busses resulted in the brief deenergization of MCC-L which results in a lockout of the feeder from MCC-L to PMIS for 15 minutes following reenergization. The PMIS 125VDC battery would assume the load via the inverter to power PMIS.

<u>Answer source:</u> 2.2.63, p. 10, step 1.2.1

Distractors:

a. PMIS would remain energized via the battery and inverter.
b. PMIS would remain energized via the battery and inverter, MDP-1 would automatically supply PMIS only if the inverter output failed.
d. MCC-L is locked out for 15 minutes following it's reenergization.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	1099	01	03/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001090C	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Automatic bus transfer

Related References	
2.2.18	4160V Auxiliary Power Distribution System

Related Skills (K/A)	
295003.AA2.01	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.10 / 43.5 / 45.13) Cause of partial or complete loss of A.C. power. (3.4/3.7)

QUESTION: 3 1099 (1 point(s))

Given the following conditions:

- ➢ The reactor is shutdown
- ➢ 4160V bus 1F has just been transferred **from** the Emergency Transformer **to** Bus 1A (1FA has just been closed)
- ➢ DG1 is running unloaded
- ➢ Breaker 1FS control switch is still in the NORMAL AFTER CLOSE position

How will the electrical system respond to a loss of the Startup Transformer at this time?

a. Breaker 1FS will close **immediately**, regardless of how long Bus 1F has been de-energized.

b. Breaker 1FS will close 12.5 seconds after the loss of Bus 1F voltage occurred.

c. DG-1 will supply 4160V Bus 1F **immediately**, regardless of how long Bus 1F has been de-energized.

d. DG-1 will supply 4160V Bus 1F after the loss of voltage has existed on Bus 1F for at least 10 seconds.

ANSWER: 3 1099

d. DG-1 will supply 4160V Bus 1F after the loss of voltage has existed on Bus 1F for at least 10 seconds.

Due to 1FS being in the NORMAL AFTER CLOSE position it will not automatically reclose. Therefore the DG will be required to supply the bus. The DG breaker always waits at least 10 seconds with the loss of voltage relay energized before automatically closing.

Answer source: 2.2.20, p. 29, step 2.7.1.5

Distractors:

a,b Incorrect because 1FS is in the NORMAL AFTER CLOSE position.
c. Incorrect as the diesel always has an at least 10 second delay before it automatically will close onto bus 1F or 1G.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	14035	00	03/31/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES

Related Lessons
COR0011302 OPS MAIN GENERATOR AND AUXILIARIES

Related Objectives	
COR0011302001060D	Describe the Main Generator and Auxiliaries design features and/or interlocks that provide for the following: Generator voltage regulation
COR0011302001080I	Predict the consequences of the following on the Main Generator and Auxiliaries: Grid instabilities
COR0011302001140D	Briefly explain the following concepts as they apply to the Main Generator: Reactive load

Related References	
2.2.14	22 KV Electrical System

Related Skills (K/A)	
245000.A4.14	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Generator megavar output (2.5/2.5)

QUESTION: 4 14035 (1 point(s))

The plant was operating at power with the following conditions present:

➢ Generator Load	765 Mwe
➢ Generator Reactive Load	-55 MVAR
➢ Generator Voltage Regulator	In Automatic control

Changing load conditions on the grid result in a slow increase in 345 KV voltage. Which of the following describes the **initial** effect of the changing voltage on main generator MVARs and field current?

MVARs become . . .

a. less negative and field amperage increases.

b. less negative and field amperage decreases.

c. more negative and field amperage increases.

d. more negative and field amperage decreases.

ANSWER: 4 14035

d. more negative and field amperage decreases.

With the increase in grid voltage and the voltage setpoint on the generator voltage regulator being held constant, the generator reactive load becomes more negative. The generator automatic voltage regulator will sense the increase in generator terminal voltage (due to the increase in 345 voltage) and reduces field amps to control voltage at the setpoint.

Answer source: 5.3GRID, p. 7, step 5.12, electrical theory

Distractors:

a. MVARs do not become less negative and field amperage does not increase.
b. MVARs do not become less negative.
c. Although MVARs do become more negative field amperage does not increase.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	14670	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0011402, OPS MAIN TURBINE

Related Lessons
COR0011402 OPS MAIN TURBINE

Related Objectives	
COR0011402001120A	Briefly describe the following concepts as they apply to Main Turbine and Auxiliaries: Feedwater heaters and Extraction Steam system operation

Related References	
2.2.29	Procedure 2.2.29, Feedwater Heaters And Extraction Steam System
2.2.77	Procedure 2.2.77, Turbine Generator

Related Skills (K/A)	
295005.AA2.03	Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.10 / 43.5 / 45.13) Turbine valve position (3.1/3.1)

QUESTION: 5 14670 (1 point(s))

The plant is operating at 90% power when the Main Generator trips.

Which of the following valves automatically **OPEN**?

a. Reheat stop valves.

b. Extraction steam dump valves.

c. Extraction Steam Non-Return valves.

d. Reactor Feed Pump Turbine low pressure steam supply valve.

ANSWER: 5 14670

b. Extraction steam dump valves.

Answer source: 2.2.29, p. 19, step 2.1

Distractors:

a, c, and d close as a result of a turbine trip.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	19124	01	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0011802, OPS Radiation Monitoring

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT COR0011802 OPS Radiation Monitoring

Related Objectives	
COR0022802001080A	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation
COR0022802001130A	Given plant conditions, determine if any of the following should occur: SGT automatic initiation
COR0011802001120E	Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Reactor Building Ventilation Isolation

Related References	
4.7.5	Procedure 4.7.5, Reactor Building Vent Exhaust Radiation Monitoring System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
295034.EK1.02	Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.8 to 41.10) Radiation releases (4.1/4.4*)

QUESTION: 6 19124 (1 point(s))

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

- ➢ RMP-RM-452A: 14 mrem/hr
- ➢ RMP-RM-452B: 7 mrem/hr
- ➢ RMP-RM-452C: 11 mrem/hr
- ➢ RMP-RM-452D: 13 mrem/hr

NO group isolations or automatic initiations occur.

What actions are required (if any) and why?
(Note: Use *actual* setpoints in your evaluation.)

a. **NO** actions are required because **neither** ***DIVISION*** logic has actuated.

b. **NO** actions are required because **only** the ***DIVISION I*** logic has actuated.

c. Manually start **only** "A" SGT train because **only** the ***DIVISION I*** logic has actuated.

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

ANSWER: 6 19124

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

If RMP-RM-452A or C <u>AND</u> RMP-RM-452B or D exceed 10 mrem/hr, Reactor Building isolates, and both SGT systems start. Per 2.0.3 "Operators shall validate automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals"

<u>Answer source:</u> 4.7.5, pp. 5 & 6, steps 1.2.3, 1.2.4, 1.2.5, & 1.3.1.1

Distractors:

a,b,c Both Divisions should have actuated. The reactor building should have isolated and both SGT trains should have started.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	5084	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0012002, OPS Reactor Water Cleanup

Related Lessons
COR0012002 OPS Reactor Water Cleanup

Related Objectives	
COR0012002001090D	Describe the RWCU design features and/or interlocks that provide for the following: Piping over-pressurization protection
COR0012002001130G	Given a RWCU component manipulation, predict and explain the changes in the following parameters: RWCU system pressure

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
2.1.31	Ability to locate control room switches / controls and indications and to determine that they are reflecting the desired plant lineup. (CFR: 45.12) (4.2/3.9)

QUESTION: 7 5084 (1 point(s))

The plant was operating at power when a RWCU isolation (Group 3) occurred.

What change in RWCU system lineup is designed to prevent overpressurization of Reactor Water Cleanup (RWCU) System Piping?

a. Return Isolation Valve, MO-68 is cracked open.

b. Blowdown Flow Control Valve PCV-55 is closed.

c. Demin Suction Bypass Valve MO-74 is cracked open.

d. Drain Valve to Radwaste System MO-57 and Drain Valve to the Condenser MO-56 are both cracked open.

ANSWER: 7 5084

c. Demin Suction Bypass Valve MO-74 is cracked open.

Following a RWCU isolation Procedure 2.1.22 requires that MO-74 be cracked open to prevent overpressurization by mini-purge. CRD purge of RWCU Pump seals can overpressurize the pump and piping following closure of MO-15 or MO-18. Opening MO-74 provides a path for CRD flow around the demins to the Reactor Vessel.

Answer source: 2.1.22, p. 10, step 6.4

Distractors:

a. MO-68 should already be open and this valve alone would not provide overpressure protection from mini-purge following isolation because a path around the now out of service demineralizers is required.
b. This valve should already be closed, in addition its closure would do nothing to prevent overpressurization of the RWCU piping. FCV-55 closes to protect downstream piping from high pressure or upstream piping from low pressure.
d. These valves should not be opened simultaneously as this could result in a loss of vacuum.

Source: ***Modified from 5084***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	14043	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001020D	Describe the interrelationships between the TEC system and the following: Control Room HVAC

Related References	
2.2.76	Procedure 2.2.76, Turbine Equipment Cooling Water System

Related Skills (K/A)	
290003.K1.05	Knowledge of the physical connections and/or cause- effect relationships between CONTROL ROOM HVAC and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Component cooling water systems (2.8/3.0)

QUESTION: 8 14043 (1 point(s))

What provides the normal and the backup cooling water to the Control Room Air conditioner?

The normal supply is from the . . .

a. Turbine Equipment Cooling (TEC) system and the backup supply is from the Service Water (SW) system.

b. Reactor Equipment Cooling (REC) system and the backup supply is from the Service Water (SW) system.

c. Turbine Equipment Cooling (TEC) system and the backup supply is from the Reactor Equipment Cooling (REC) system.

d. Reactor Equipment Cooling (REC) system and the backup supply is from the Turbine Equipment Cooling (TEC) system.

ANSWER: 8 14043

a. Turbine Equipment Cooling (TEC) system and the backup supply is from the Service Water (SW) system.

TEC supplies cooling to the Control Room Air Conditioner and can be supplied from SW by manually positioning local valves.

Answer source: 2.2.76, p. 33, step 1.2.2.7.

Distractors:

b. REC is not capable of providing the normal supply the Control Room AC unit.
c. REC is not capable of supplying the backup cooling to the Contol Room AC unit.
d. REC is not capable of providing the normal supply the Control Room AC unit and TEC is the normal supply.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	14045	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0013001, HWC Gas Generation System

Related Lessons
COR0013001 HWC Gas Generation System

Related Objectives	
COR0013001001040D	Identify the reason/function of the following systems interface and general physical location of the interface with the HWC system: Offgas System

Related References	
2.2.98	Hydrogen/Oxygen Generation System

Related Skills (K/A)	
272000.K5.01	Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: (CFR: 41.7 / 45.4) Hydrogen injection operation's effect on process radiation indications: Plant-Specific (3.2/3.5)

QUESTION: 9 14045 (1 point(s))

The plant is operating at 100% power with the hydrogen injection in service when OWC INJECTION SYS SHUTDOWN, A-3/F-4 alarms. The Control Room operator places the OWC INJECTION SYS ENABLE SWITCH to SHUTDOWN and verifies the that the green (Shutdown) light is on.

How does this affect ERP radioactive release rate and Main Steam Line (MSL) radiation level?

ERP release rate . . .

a. increases and MSL radiation levels increase.

b. decreases and MSL radiation level decrease.

c. is unchanged and MSL radiation level decrease.

d. is unchanged and MSL radiation level is unchanged.

ANSWER: 9 14045

c. is unchanged and MSL radiation level decrease.

The indications, annunciator and operator action indicate a loss of hydrogen injection. The loss of the hydrogen injection results in a shift of the ratio of N-16 as ammonia or ammoniun to nitrate or nitrite anion forms. This results in less carryover of N-16 out the main steam lines and a reduction in MSL radiation levels. Since N-16 has a short half life this change in carryover does not effect the release rate out the ERP.

Answer source: COR012-03-01, p. 4

"What is going to happen if Cooper starts adding hydrogen to the reactor water? According to the previous paragraph, if the nitrogen reacts with hydrogen, ammonia if formed. With a lot more hydrogen in the Reactor to combine with, the nitrogen will combine with it. Consequently there will be a lot more nitrogen-16 going over to the turbine and dose rates will be much higher."

Distractors:

a. ERP release rate does not increase.
b. ERP release rate does not decrease.
d. MSL radiation levels decrease.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	2931	02	03/18/2003	06/13/2003	Licensed Operator	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001050F	Describe the interrelationship between the Primary Containment system and the following: Plant Air
COR0020302001120F	Describe the Containment design features and/or interlocks that provide for the following: Reactor building to Torus D/P

Related References	
3.6.1.7	Reactor building-to-suppression chamber vacuum breakers
COR0020302	Containment

Related Skills (K/A)	
295019.AK2.09	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8) Containment. (3.3/3.3)

QUESTION: 10 2931 (1 point(s))

Drywell sprays are in service to support EOP actions when a complete loss of instrument air (including nitrogen) occurs. Due to a logic failure, the drywell spray valves (RHR-MO-26A and RHR-MO-31A) have been opened manually using the local handwheels.

Is the torus protected from exceeding design negative pressure under these conditions and why/why not?

a. Yes, all reactor building-to-torus vacuum breakers are motor-operated.

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

c. No, the reactor building-to-torus vacuum breakers fail closed on a loss of air.

d. No, the reactor building-to-torus vacuum breakers are not designed to facilitate this amount of flow.

ANSWER: 10 2931

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

Answer source: COR002-03-02, p. 20

Distractors:

a. One of the vacuum breakers is pneumatically operated.
c. The MOV doesn't fail anywhere on loss of air. The AOV fails open.
d. The vacuum breakers are sized to facilitate this flow.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	10081	02	03/21/2003	06/13/2003	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	COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001080D	State the electrical power supplies to the following: N2 solenoid valve
COR0020302001240B	Predict the consequences of a malfunction of the following on PCIS: DC electrical.
COR0020302001230C	Predict the consequences of a malfunction of the following on the Primary containment: Containment atmospheric control/nitrogen make-up.

Related References	
2.2.60	Procedure 2.2.60, Primary Containment Cooling And Nitrogen Inerting System
2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1
2.2.59	Procedure 2.2.59, Plant Air System

Related Skills (K/A)	
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) (3.9/4.0)

QUESTION: 11 10081 (1 point(s))

The plant is at 100% power with the following conditions:

- Annuciator 9-3-1/G-2, NITROGEN SOLENOID DE-ENERGIZED is in alarm
- 125 VDC Panel AA2 is de-energized

Which of the following would be the ***quickest*** action that will restore pressure to the drywell pneumatic header?
(NOTE: The choices are listed from QUICKEST to LONGEST order.)

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

b. Open the Reactor building drywell supply air valve (IA-V-571) above the Southeast Hydraulic Control Units.

c. Open RR-SPV-740 AND RR-SPV-741 SUPPLY SHUTOFF (IA-1672) near RWCU precoat pump.

d. Hook up the nitrogen bottles that are stored in a rack near the header.

ANSWER: 11 10081

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

Answer source: 2.3_9-3-1, p. 74,
2.2.59 p. 25, step 1.2.11

Distractors:

b. This would take a person some time to manually open the valve (faster than bottles).
c. These valves are already open and would not restore drywell pneumatics.
d. This would take at least one person, some tools and time.

2.1.23 Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) as it applies to: 223002 PCIS/Nuclear Steam Supply Shutoff

<u>Source:</u> ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	5155	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020402, CONTROL ROD DRIVE HYDRAULICS

Related Lessons
COR0020402 CONTROL ROD DRIVE HYDRAULICS

Related Objectives	
COR0020402001140A	State the electrical power supply to the following CRDH components: CRDH pumps motors.

Related References	
5.3EMPWR	EMERGENCY POWER
2.2.8	Procedure 2.2.8, Control Rod Drive System
2.2.8A	Procedure 2.2.8A, Control Rod Drive Hydraulic System Valve Checklist

Related Skills (K/A)	
201001.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: (CFR: 41.7 / 45.7) A.C. power (3.3/3.3)

QUESTION: 12 5155 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 65% power
- ➢ The "B" Control Rod Drive (CRD) Pump is running
- ➢ 480 VAC Critical Switchgear Bus 1G trips

What is the status of "B" CRD pump and Drive Water DP?

a. The "B" Pump is running.
Drive Water DP is unaffected.

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

c. The "B" Pump is stopped.
Drive Water DP will decay away over the next several minutes.

d. The "B" Pump is running.
The Drive Header Pressure Control Valve has lost power.

ANSWER: 12 5155

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

480 VAC Bus 1G provides power to CRD Pump "B" so Pump "B" is stopped. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.

Answer source: 2.2.8A, p. 10

Distractors:

a. The "B" pump is powered by 480 VAC Bus 1G and will trip.
b. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.
d. The "B" pump is powered by 480 VAC Bus 1G and will trip.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	14040	00	03/13/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020602, CORE SPRAY

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001080H	Given a Core Spray component manipulation, predict and explain the changes in the following: System lineup
COR0020602001120A	Given plant conditions, determine if any of the following Core Spray Actions should occur: System initiation.
COR0020602001120D	Given plant conditions, determine if any of the following Core Spray Actions should occur: Valve reposition.
COR0020602001050E	Describe the Core Spray system design features and/or interlocks that provide for the following: Pump minimum flow

Related References	
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
209001.A3.03	Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: (CFR: 41.7 / 45.7) System pressure (3.5/3.5)

QUESTION: 13 14040 (1 point(s))

The plant was operating at power with the "A" Core Spray subsystem in full flow test at 5000 gpm. An accident occurs that results in increasing drywell pressure and lowering reactor water level and lowering reactor pressure. The following plant conditions exist:

- ➢ Drywell pressure 11 psig (rising)
- ➢ Reactor water level -21" (wide range, lowering)
- ➢ Reactor pressure 375 psig (lowering)

What is the pressure response of the A Core Spray system ***at this time***?

Core spray system <u>pressure</u> . . .

a. remains the same.

b. increases to pump shut-off head.

c. decreases to <u>just above</u> reactor pressure.

d. increases to <u>just below</u> pump shut-off head.

ANSWER: 13 14040

d. increases to <u>just below</u> pump shut-off head.

An initiation signal is present for the Core Spray System. The CS test valve would receive a close signal resulting in a significant reduction in flow and since reactor pressure remains above the shut-off head for the pumps flow would be reduced to the point that the minimum flow valve would open. Core Spray system pressure would then be just below pump shut-off head.

<u>Answer source:</u> 2.2.9 p. 20

Distractors:

a. Core Spray pressure would not remain the same because system flow rate would become significantly reduced when the initiation signal occurred and the test line isolated.
b. Core Spray system pressure would be below shut-off head because of the flow that exists through the minimum flow valve.
c. Core Spray pressure increases.

<u>Source:</u> *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	14050	00	03/28/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020702, OPS DC ELECTRICAL DISTRIBUTION

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001090A	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Manual/automatic transfers of control
COR0020702001090B	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Breaker interlocks, permissives, bypasses and crossties

Related References	
2.2.25.2	125 VDC ELECTRICAL SYSTEM (DIV 2)

Related Skills (K/A)	
263000.K4.02	Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Breaker interlocks, permissives, bypasses and cross ties: Plant-Specific (3.1/3.5)

QUESTION: 14 14050 (1 point(s))

The plant is operating at rated power when the normal power supply to the 125VDC HPCI Starter Rack is lost.

What interlocks exist for transfer of the HPCI Starter Rack to its alternate supply?

Inadvertent transfer of the 125VDC HPCI Starter Rack from its normal to its alternate supply is prevented by transfer switch design . . .

a. **only**.

b. **AND** the alternate supply breaker is locked open **only**.

c. **AND** a mechanical interlock prevents closing both supply breakers simultaneously **only**.

d. **AND** the alternate supply breaker is locked open **AND** a mechanical interlock prevents closing both supply breakers simultaneously.

ANSWER: 14 14050

b. **AND** the alternate supply breaker is locked open **only**.

Answer source: 2.2.25.2 pp. 35 & 36, sections 38.3, 38.4, 38.5 & 38.6

Distractors:

a. The alternate supply is locked open.
c. The alternate supply is locked open and both switches are closed at the same time during a transfer.
d. Both switches are closed at the same time during a transfer.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	19090	01	03/28/2003	06/13/2003	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
Integrated Plant	COR0020702, OPS DC ELECTRICAL DISTRIBUTION

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives	
COR0020702001080L	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: SRMs
COR0020702001080R	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Radiation Monitoring systems
COR0021202001070B	Predict the consequences of a loss or malfunction of the following would have on the IRM system: 24/48 VDC
COR0020702001080J	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor Protection system

Related References	
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.2.26	Procedure 2.2.26, 24 VDC Electrical System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
215003.A3.03	Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7 / 45.7) RPS status (3.7/3.6)

QUESTION: 15 19090 (1 point(s))

The station is in MODE 2 withdrawing control rods in an approach to criticality during a startup. The following equipment simultaneously trips:
(NOTE: Other equipment also trips but is not required to assess conditions.)

- IRM "A", "C", "E" and "G"
- SRM "A" and "C"
- Off-Gas Radiation monitor "A"
- Reactor Building Vent Radiation monitors "A" and "C"
- Control rods remain at their pre-transient position
- **NO** group isolations have occurred

What occurred and what actions (if any) are required?

a. A loss of RPSPP "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

b. A loss of RPSPP "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

c. A loss of 24 VDC "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

ANSWER: 15 19090

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

The loss of 24 vdc will cause all these instruments to become inoperative. No scram or group isolation will occur due to this single power loss.

Answer source: 2.2.22, p. 9 (RPS loss distractors),
2.2.26, step 2.2.1

Distractors:

a. RPS power loss would not cause the loss of IRMs/SRMs. No ATWS or group isolation failure has occurred.
b. RPS power loss would not cause the loss of IRMs/SRMs.

c. No ATWS or group isolation failure has occurred.

Source: ***Direct From Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	1507	02	03/31/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020802, Diesel Generators

Related Lessons
COR0020802 DIESEL GENERATORS COR0010102 AC Electrical Distribution

Related Objectives	
COR0020802001090E	Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Load Shedding and Sequencing
COR0010102001130B	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of coolant accident
COR0010102001130C	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of off-site power

Related References	
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation
COR0020802	Diesel Generators

Related Skills (K/A)	
264000.K5.06	Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3) Load sequencing (3.4/3.5)

QUESTION: 16 1507 (1 point(s))

The plant was operating at rated power when a loss of all off-site power occurred coincident with a large recirculation suction line break.

What will be the sequential loading of emergency buses?
(T = DG Output Breaker Closure)

a. T+0 the Core Spray pump starst;
T+5 seconds the first RHR Pump starts;
T+10 seconds the second RHR pump and SGT start.

b. T+0 the first RHR pump starts;
T+5 seconds the Core Spray pump and SGT start;
T+10 seconds the second RHR pump starts.

c. T+0 the first RHR pump and SGT starts;
T+5 seconds the Core Spray pump starts;
T+10 seconds the second RHR pump starts.

d. T+0 the first RHR pump and SGT starts;
T+5 seconds the second RHR pump starts;
T+10 seconds the Core Spray pump starts.

ANSWER: 16 1507

d. T+0 the first RHR pump and SGT starts;
T+5 seconds the second RHR pump starts;
T+10 seconds the Core Spray pump starts.

Answer source: COR002-08-02, p. 65, Table 1

Distractors:

a. RHR pump starts first and second, CS starts last. SGT starts at T=0.
b. RHR pump starts first and second, CS starts last. SGT starts at T=0.
c. RHR pump starts first and second, CS starts last.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	14032	00	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021102, OPS High Pressure Coolant Injection (HPCI)

Related Lessons
COR0021102 OPS High Pressure Coolant Injection (HPCI)

Related Objectives	
COR0021102001080N	Describe the HPCI design features and/or interlocks that provide for following: Pump minimum flow
COR0021102001100H	Predict the consequences of the following on the HPCI system: Low ECST level
COR0021102001080M	Describe the HPCI design features and/or interlocks that provide for following: Protection against draining the CST to the torus

Related References	
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
206000.A4.07	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Condensate storage tank level: BWR-2,3,4 (3.5/3.5)

QUESTION: 17 14032 (1 point(s))

The plant was operating at power with the HPCI system in full flow test per 6.HPCI.103 when annunciator 9-3-2/A-4, HPCI SUCTION TRANSFER alarms. ECST level is 22".

What is the status/alignment of HPCI several minutes later?
(Assume the operator takes no action.)

HPCI suction valves are aligned to the suppression pool, the Minimum Flow Valve (MO-25) is _________ and the Pump Test Return Line Isolation Valves (MO-21 & 24) are _________.

a. open closed

b. closed closed

c. open open

d. closed open

ANSWER: 17 14032

a. open closed

The low ECST level has initiated a swap of the of the HPCI suction valves. With the swap over to suction from the suppression pool the pump test return isolation valves automatically close. Now the HPCI system is without a discharge path the minimum flow valve opens due to low flow.

Answer source: 2.2.33 p. 15, steps 2.1.1.8 (MO-25), 2.1.1.9 (MO-21) & 2.1.1.10 (MO-24)

Distractors:

b. The minimum flow valve is open.
c. The Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.
d. The minimum flow valve is open and the Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	19051	01	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021102, OPS High Pressure Coolant Injection (HPCI)

Related Lessons	
COR0021102	OPS High Pressure Coolant Injection (HPCI)
SKL0124211	HIGH PRESSURE COOLANT INJECTION

Related Objectives	
COR0021102001120A	Given plant conditions, determine if the following HPCI actions should occur: System initiation
SKL012421100A030J	Given plant conditions, predict changes in the following HPCI system components/parameters: Turbine speed
SKL012421100B0600	Comply with all related HPCI system limits and precautions.
COR0021102001100V	Predict the consequences of the following on the HPCI system: High reactor water level
SKL012421100A0200	Explain the HPCI system limitations and precautions as stated in the SOP 2.2.33 and SOP 2.2.33.1.

Related References	
791E271	HPCI System Elementary Diagram
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.4/3.8)

QUESTION: 18 19051 (1 point(s))

The plant was operating at power when a loss of off-site power occurred. The reactor scrammed and HPCI started on low reactor water level. Reactor water level quickly recovered and the HPCI turbine tripped on high RPV water level. The following plant conditions were present:

- ➢ Reactor water level 45" (NR) (lowering slowly)
- ➢ Reactor pressure 850 psig (rising slowly)
- ➢ Drywell pressure 2.2 psig (rising slowly)

What is/are the **MINIMUM** action(s) required to restart HPCI ***at this time***?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

b. Momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

c. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

d. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

ANSWER: 18 19051

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

Per 2.2.33.1:

CAUTION - If HPCI initiation signal cannot be reset, HPCI System will automatically start when RX HI WTR LEVEL SIGNAL RESET pushbutton is depressed and vessel level is ≤ +54".

During the transient drywell pressure has risen to greater than the initiation setpoint for HPCI. Since an automatic initiation signal is present, if the operator depresses the Reactor Hi Water Level Signal Reset pushbutton the system will reinitiate.

Answer source: 2.2.33.1, p. 8
2.2.33, p. 20, step 2.2.6

Distractors
b, c, d - only the high level trip reset need be depressed.

2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) as it applies to: 295008 High Reactor Water Level / 2

<u>Source:</u> ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	16513	01	02/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021402, Digital Electro-Hydraulic Control

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040B	Describe how the DEH control system operates to control the following: Reactor pressure
COR0020902001070B	Given a specific DEH Control system malfunction, determine the effect on any of the following: Reactor pressure

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

Related Skills (K/A)	
241000.K4.01	Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Reactor pressure control (3.8/3.8)

QUESTION: 19 16513 (1 point(s))

The plant is operating at 100% power when the in-service DEH pressure controller fails such that controller output INCREASES slowly.

Assuming NO operator action is taken, what is the plant response?

a. The reactor scram due to high reactor pressure.

b. The MSIVs isolate due to low reactor pressure.

c. Turbine throttle pressure will be controlled about 4 psig LOWER than before the failure.

d. Turbine throttle pressure will be controlled about 4 psig HIGHER than before the failure.

ANSWER: 19 16513

b. The MSIVs isolate due to low reactor pressure.

As the controller output increases the turbine governor valves would open in response to the controller output. The opening of the valves would reduce reactor pressure and result in a MSIV closure due to low MSL pressure with the mode switch in RUN.

Answer source: COR002-09-02, p. 49, section "c."

Distractors:

a: Reactor pressure will lower as controller output signals the TCVs to OPEN.
c: The backup pressure regulator is set for a pressure 4 psi higher.
d: Reactor pressure will lower as controller output signals the TCVs to OPEN.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	1058	01	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021502, NUCLEAR BOILER INSTRUMENTATION

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001040H	Briefly describe the following concepts as they apply to NBI: Recirculation flow effects on level indicators

Related References	
COR0021502	Nuclear Boiler Instrumentation

Related Skills (K/A)	
295001.AA1.07	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.7 / 45.6) Nuclear boiler instrumentation system. (3.1/3.2)

QUESTION: 20 1058 (1 point(s))

The plant is operating at 100% power when both Reactor Recirculation pumps slowly run back to minimum speed. No operator action is taken.

What is the difference between ACTUAL and INDICATED Wide Range Reactor Water Level ***prior*** to the power reduction AND what is the expected change in that difference ***during*** the power reduction?

Prior to the power reduction, *actual* downcomer level is ___________ than *indicated* downcomer level AND the difference will get ___________ during the power reduction.

a. lower larger

b. higher larger

c. lower smaller

d. higher smaller

ANSWER: 20 1058

d. higher smaller

Due to the velocity affects of flow in the annulus, the variable leg will sense a lower pressure than is exerted by the height of water alone. This is seen as a lower indicated level. At higher recirc flows, higher velocities cause a greater difference between Wide Range indicated and actual levels. The difference between indicated and actual levels can range from 4-18"

Answer source: COR002-15-02, p. 42 & 43, section 2.c, d, f, g, & h.

Distractors:

a. Actual level is higher than indicated level. The difference will get smaller as power is reduced.
b. The difference will get smaller as power is reduced.
c. Actual level is higher than indicated level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	5425	01	03/24/2003	06/13/2003	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	COR0021602, OPS NUCLEAR PRESSURE RELIEF

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001050J	Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: Safety/Relief operating signals
COR0021602001030J	Describe the interrelationships between the Nuclear Pressure Relief system and the following: RPS (low-low set initiation)

Related References	
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
239002.K6.03	Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: (CFR: 41.7 / 45.7) A.C. power: Plant-Specific (2.7*/2.9*)

QUESTION: 21 5425 (1 point(s))

Given the following:

- ➢ The plant is operating at 100% power at Beginning of Life (BOL)
- ➢ A loss of off-site power occurs
- ➢ All control rods fully inserted
- ➢ Pressure rises to 1090 psig, **THEN** lowers to 875 psig
- ➢ 20 minutes later, pressure is cycling between 1015 **AND** 875 psig

What is the status of Low Low Set (LLS) and why?
(NOTE: **NO** operator action is taken.)

a. LLS is controlling pressure. LLS logic has no AC powered inputs or components.

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

c. LLS is **NOT** controlling pressure. With RPS power unavailable, the SRVs must operate on mechanical relief setpoint to control pressure.

d. LLS is **NOT** controlling pressure. A fault must exist in the LLS logic as all conditions are present for LLS to automatically control pressure.

ANSWER: 21 5425

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

Answer source: COR002-16-02, p. 24 & p. 24, p. 43 section E,
COR002-16-02 Figure 6

Distractors:

a. LLS logic is armed by RPS high pressure signal.
c. LLS logic will arm on high pressure with no RPS power available.
d. LLS is controlling pressure.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	5608	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021602, OPS NUCLEAR PRESSURE RELIEF

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001020A	State the electrical power supply to the following NPR components: ADS logic
COR0021602001080F	Predict the consequences a malfunction of the following would have on the NPR system: D.C. power

Related References	
791E253	Automatic Blowdown System
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
218000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) ADS logic (3.1*/3.3*)

QUESTION: 22 5608 (1 point(s))
An accident has occurred, resulting in the following conditions:

➢	Reactor pressure	720 psig (lowering)
➢	RPV water level	-120" (WR stable)
➢	Drywell pressure	6.2 psig (rising)
➢	125 VDC panel AA2	De-energized

If present conditions continue, how will ADS respond?

ADS valves will . . .

a. ***fail to open*** due to loss of logic power.

b. ***fail to open*** due to RPV water level conditions not met.

c. be opened by the B logic circuit powered from its ***normal*** power source.

d. be opened by both logic circuits powered from their ***alternate*** power sources.

ANSWER: 22 5608

c. be opened by the B logic circuit powered from its ***normal*** power source.

Answer source: COR002-16-02, p. 21, & p. 22, section 3, p. 41
COR002-16-02 Figures 4 & 5

Distractors:

a. ADS will initiate powered from BB2.
b. ADS will initiate.
d. ADS "A" has no alternate source and is de-energized.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	14679	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM

Related Lessons
COR0021702 PLANT MANAGEMENT INFORMATION SYSTEM

Related Objectives	
COR0021702001070A	Given a specific PMIS malfunction, determine the effect on any of the following: On Demand print out.

Related References	
2.6.3	Procedure 2.6.3, COMPUTER SYSTEMS OPERATION AND OUTAGE RECOVERY
2.4COMP	Computer Malfunction

Related Skills (K/A)	
2.1.19	Ability to use plant computer to obtain and evaluate parametric information on system or component status. (CFR: 45.12) (3.0/3.0)

QUESTION: 23 14679 (1 point(s))

A plant shutdown is in progress with reactor power at 30% of rated and both PMIS computers in service when a loss of the primary PMIS computer occurs.

What is the impact of these conditions on plant operation?

Official Cases are . . .

a. available and, if the shutdown were to continue, RWM would be available.

b. available and, if the shutdown were to continue, RWM would be **UN**available.

c. **UN**available and, if the shutdown were to continue, RWM would be available.

d. **UN**available and, if the shutdown were to continue, RWM would be **UN**available.

ANSWER: 23 14679

a. available and, if the shutdown were to continue, RWM would be available.

When both PMIS computers are unavailable, monitoring functions (Computer Edits) 3D Monicore Official Cases, process parameter alarm monitoring and many other important functions are lost; however, in this instance, with the backup computer available, a Loss of primary computer will result in automatic fail-over to the backup computer, so there is no immediate impact on plant operation, all computer functions are available.

Answer source: COR002-17-02, p. 9, section B.1

Distractors:

b. The backup computer is available so the RWM remains available on the subsequent shutdown.
c. The backup computer is available so computer on-demand printouts remain available.
d. The backup computer is available so computer on-demand printouts remain available as does the RWM on the susequent shutdown.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	14451	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling (RCIC)

Related Objectives	
COR0021802001120D	Given plant conditions, determine if the following RCIC actions should occur: Minimum flow valve position change
COR0021802001120A	Given plant conditions, determine if the following RCIC actions should occur: RCIC system initiation
COR0021802001100O	Predict the consequences of the following on the RCIC system: RCIC Turbine control system failure
COR0021802001120E	Given plant conditions, determine if the following RCIC actions should occur: RCIC turbine trip

Related References	
2.2.67	Procedure 2.2.67, Reactor Core Isolation Cooling System

Related Skills (K/A)	
217000.K5.02	Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.5 / 45.3) Flow indication (3.1/3.1)

QUESTION: 24 14451 (1 point(s))

The Reactor Core Isolation Cooling (RCIC) flow transmitter has failed low such that it senses 0 gpm irrespective of actual RCIC flow.

What is the expected RCIC system response upon receipt of a valid initiation signal?

The RCIC turbine will start and . . .

a. run normally.

b. trip on overspeed.

c. run continuously at minimum speed.

d. run continuously at approximately 4500 rpm.

ANSWER: 24 14451

d. run continuously at approximately 4500 rpm.

Loss of flow signal input to the flow controller results in a maximum speed demand signal. Since the output of the control box is limited to 50 milliamps, the turbine speed will top out at approximately 4500 rpm.

Answer source: COR002-18-02, p. 56, section 2

Distractors:

a. RCIC will not run normally.
b. The ramp generator is still functional on startup. RCIC RPM will not exceed 4500 when controller output is at 100%. Overspeed occurs at 5625 RPM.
c. RCIC will not run at mimimum speed.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	14027	00	02/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Generic Fundamentals	ACD0070307, Thermal Hydraulics (GP)

Related Lessons
ACD0070306 Heat Transfer & Heat Exchangers (GP) ACD0070307 Thermal Hydraulics (GP)

Related Objectives	
ACD00703020010800	Apply saturated and superheated steam tables in solving liquid-vapor problems.
ACD00703060010500	Solve heat flux and heat transfer rate problems.
ACD0060507001310C	Explain the relationship between decay heat generation and: time since reactor shutdown

Related References	
ACD0060507	Reactor Operational Physics

Related Skills (K/A)	
295007.AK1.02	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Decay heat generation (3.1/3.4)

QUESTION: 25 14027 (1 point(s))

The plant has been shutdown for several days following a long operating cycle. Shutdown Cooling has been placed in service and a cooldown established. The following conditions are present:

- Reactor pressure is 15 psig
- Reactor vessel inventory is 350,000 lbm
- Decay heat rate is 0.6% of rated thermal power
- Ambient heat loss is 7.5 MWt
- Reactor coolant specific heat capacity is 1.08 BTU/lbm°F

A station blackout then occurs.

What reactor pressure will exist **two** (2) hours following this loss of all AC power?
(Assume that reactor inventory and ambient losses remain constant for the entire two hours.)

a. 83 psig

b. 90 psig

c. 165 psig

d. 180 psig

ANSWER: 25 14027

c. 165 psig

The thermal power of the reactor is 14.3 Mwt (.006*2381=14.3Mwt). The thermal power that is absorbed in the coolant is 14.3Mwt - 7.5 Mwt (ambient loss) =6.8Mwt. 6.8 Mwt is converted to BTU/hr by multiplying by 3.41E6. This yields 2.31E7 BTU/hr. Since the question asks for the conditions 2 hours after the loss of AC power this heat rate continues for 2 hours so 2.31E7BTU/hr is multiplied by 2 hrs to yield the total BTUs absorbed by the coolant or 4.62E7 BTUs. Now the known values can be substituted into the following equation to solve for the final temperature. $Q=Mc_p\Delta T$ The final temperature is calculated to be 373°F. Now steam tables are used to find the saturation pressure for that temperature (180 psia). The value is then converted to psig and the final answer of 162 psig. As the RPV is an enclosed volume under these conditions, addition of energy to the mass of water will result in a temperature change and very little phase change. The amount of energy lost to the latent heat of vaporization is insignificant when compared to the total enthalpy of the water in the RPV.

Answer source: Steam Tables, Generic Fundamentals

Distractors:

a. Would only be obtained if the candidate failed to account for two hours after the loss of AC power.
b. Would be obtained if the candidate failed to account two hours since the loss of AC power and failed to convert that answer to psig.
d. Would be only be obtained if the candidate failed to convert the final answer to psig.

<u>Source</u>: ***New***

<u>Provide to Candidate:</u> Calculator, Steam Tables, GFES Formula sheet.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	2521	03	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021902, REACTOR EQUIPMENT COOLING

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060K	Given a specific REC malfunction, determine the efect on any of the following: Fuel Pool Cooling system

Related References	
2.2.65	Procedure 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: 26 2521 (1 point(s))

Closure of which of the following REC valves could lead to an increase in the fuel pool water temperature and in increase in airborne contamination?

a. Drywell Supply Isolation (REC-MO-702).

b. Non-Critical Header Supply (REC-MO-700).

c. Augmented Radwaste Supply (REC-MO-1329).

d. Critical Loop Return Crossover Valve (REC-MO-694).

ANSWER: 26 2521

b. Non-Critical Header Supply (REC-MO-700).

Answer source: 2.2.65, pp. 12 & 13, section 1.2

Distractors:

a. FPC is supplied by the non-critical header.
c. FPC is supplied by the non-critical header.
d. FPC is supplied by the non-critical header.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	14039	00	03/11/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021902, REACTOR EQUIPMENT COOLING

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001110A	Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation
COR0021902001110C	Given plant conditions, determine if any of the following should occur: Any REC valve automatic reposition

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System

Related Skills (K/A)	
400000.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: (CFR: 41.7 / 45.7) Motors (2.8/2.9)

QUESTION: 27 14039 (1 point(s))

The plant was at power with Reactor Equipment Cooling (REC) pumps A, B, and D running and REC pump C tagged out for maintenance. The B REC pump then tripped due to an electrical fault in the motor. REC pressure lowered to 40 psig before increasing and stabilizing two (2) minutes later at 53 psig.

Which of the following loads **CAN** be supplied with REC?

a. "A" Drywell Fan Coil Unit

b. "A" Station Air Compressor

c. "A" Control Rod Drive pump

d. Northwest Quad Fan Coil Unit.

ANSWER: 27 14039

d. Northwest Quad Fan Coil Unit.

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 overriden. This will isolate REC to the non-critical loops/components. The fan coil unit is the only load listed supplied from the critical loop.

Answer source: 2.2.65, p. 14, section 2.5
2.2.65, p. 15, sections 2.9 & 2.10

Distractors:

a. The drywell fancoil will remain isolated because REC pressure remains below the isolation setpoint.
b. REC flow to the air compressor will remain isolated because REC pressure remains below the isolation setpoint.
c. REC flow to the CRD pump will remain isolated because REC pressure remains below the isolation setpoint.

Source: Modified *Original Question 5279*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	14024	00	02/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001150E	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod drift alarm
COR0022002001010I	State the purpose of the following items related to the Reactor Manual Control System and/or the Rod Position Information System: Rod Drift Alarm Test Switch

Related References	
6.CRD.303	CONTROL ROD WITHDRAWAL/OPERABILITY TEST MODE 3, 4, AND 5

Related Skills (K/A)	
201002.A4.03	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Rod drift test switch (2.8/2.8)

QUESTION: 28 14024 (1 point(s))

What represents the **MINIMUM** action(s) required to generate a rod drift alarm?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Rod drift test switch momentarily held to test position.

b. Rod drift test switch held in the test position while control rod is inserted one notch.

c. Rod drift test switch held in the test position while a control rod is inserted and then withdrawn one notch.

d. Control rod inserted one notch and the rod drift test switch is momentarily taken to test while the amber rod settle light is energized.

ANSWER: 28 14024

b. Rod drift test switch held in the test position while control rod is inserted one notch.

Any Rod movement that leaves an even reed switch or picks up an odd reed switch with the Rod Drift Alarm Test switch in test generates a Rod Drift Alarm.

Answer source: COR002-20-02, p. 18, section 4

Distractors:

a. Just Placing the Rod Drift Alarm Test switch to TEST does not generate a rod drift alarm.

c. While this would generate a rod drift alarm, the rod need only be either inserted OR withdrawn, so these actions do not represent the minimum required by the question.

d. This action may not generate a rod drift alarm, if the even reed switch for the control rod is made up before the Rod Drift Alarm Test switch is taken to TEST, no alarm would be generated.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	1208	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022102, REACTOR PROTECTION SYSTEM

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001100K	Describe the interrelationship between the RPS and the following: Primary Containment
COR0022102001050A	Briefly describe the following concepts as they apply to RPS: Logic arrangements

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
212000.A2.09	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 conseq...: (CFR: 41.5 / 45.6) High containment/drywell pressure (4.1*/4.3*)

QUESTION: 29 1208 (1 point(s))

The plant is operating at 10% power and rated pressure during a plant startup with the following conditions:

- The "A" Reactor Protection System (RPS) MG has tripped due to a motor fault
- A loss of Drywell (DW) cooling has caused a slow but continuous rise in DW temperature and pressure
- Primary Containment parameters do not improve

What drywell pressure switch configuration will result in a full reactor scram and how is overfill of the RPV prevented?

A reactor scram will occur . . .

a. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

b. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

d. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

ANSWER: 29 1208

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

<u>Answer source:</u> COR002-21-02 p.12, section 3
2.1.5 p. 8, section 1.3

Distractors:

a. closing the drive pressure control valve will not prevent overfill.
b. Does not need to be both switches, closing the drive pressure control valve will not prevent overfill.
d. Does not need to be both switches.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	19096	00	05/22/2002	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022202, REACTOR RECIRCULATION

Related Lessons
SKL0124222 REACTOR RECIRCULATION SYSTEM COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001130F	Given plant conditions, determine if any of the following should occur: Recirculation MG set scoop tube lock.
COR0022201001060A	Given plant and/or reactor recirculation system conditions, apply the design features and/or interlocks that provide for the following: MG Set Scoop Tube Lockout
SKL012422200A030I	Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: RR pump speed

Related References	
2.2.68	Procedure 2.2.68, Reactor Recirculation System Operations

Related Skills (K/A)	
202002.A4.01	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) MG sets (3.3/3.1)

QUESTION: 30 19096 (1 point(s))

The plant is operating at 30% when the reactor operator is directed to raise power using Recirculation flow. As the controller output is raised, a momentary (1.5 seconds) loss of signal occurs from the "A" Recirculation Flow Controller. The operator continues to raise the controller output for several more seconds.

How will the "A" Recirculation MG Set be affected by this momentary loss and operator action?

a. The pump will automatically run back to ~ 22% speed.

b. A scoop tube lockup will prevent any further speed change.

c. After a 1.5 second pause, recirculation pump speed will rise for several seconds.

d. Speed will initially rise, then lower rapidly for 1.5 seconds, then rise again for several seconds.

ANSWER: 30 19096

b. A scoop tube lockup will prevent any further speed change.

Answer source: COR002-22-02, p. 35, section "d"

Distractors:

a. There is no runback.
b. Scoop tube lockup prevents any speed changes.
d. Scoop tube lockup prevents any speed changes.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	1744	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030P	Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling
COR0022302001170C	Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Related References	
2.2.69.3	Procedure 2.2.69.3, RHR Suppression Pool Cooling And Containment Spray
2.2.69	Procedure 2.2.69, Residual Heat Removal System

Related Skills (K/A)	
2.4.48	Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) (3.5/3.8)

QUESTION: 31 1744 (1 point(s))

Following a LOCA, the following conditions are present :

- Reactor pressure 700 psig (lowering slowly)
- RPV water level - 100 in (**wide range**, stable)
- Drywell press 11.0 psig (rising slowly)

What are the **MINIMUM** actions that are required in order to initiate Drywell Sprays?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

c. Place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

d. Depress Containment Spray Initiation Signal Reset pushbuttons, place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

ANSWER: 31 1744

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

Answer source: 2.2.69, pp. 35 - 36, section 2.2.10

Distractors:

a. The permissive switch must be placed in MANUAL.
c. No need to place 2/3 core height in override.
d. No need to place 2/3 core height in override or reset logic.

2.4.48 Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) as it applies to: 295024 High Drywell Pressure

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	4029	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	8	Multiple Choice	

Topic Area	Description
Systems	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001020A	State the electrical power supplies to the following: RHR pump motors
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System

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

QUESTION: 32 4029 (1 point(s))

The plant was at power with the breaker for RHR Pump 1C tagged out for maintenance when the following occur:

- ➢ A reactor coolant leak in the drywell results in a drywell pressure of 8 psig (slowly rising)
- ➢ A loss of 4160 VAC Switchgear Critical Bus 1F occurs

What RHR pumps/loops remain available and what operations be accomplished <u>from the Control Room</u>?

RHR . . .

a. Loop A with **one** pump is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

c. Loop B with **both** pumps is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

d. Loop A with **one** pump **AND** RHR Loop B with **one** pump are available for LPCI injection. Torus sprays are available from RHR loop A **ONLY**.

ANSWER: 32 4029

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

Torus sprays are available from RHR loop B **ONLY**. The loss of power to 1F results in the loss of power to RHR pumps A and B. With C pump already out of service, the only remaining pump is RHR pump D, a B loop pump. Since the B loop injection valve is DC powered LPCI remains available from B loop. Power remains available to the containment cooling valves for the B loop so torus spray is available.

<u>Answer source:</u> COR002-23-02, pp. 18 & 19, Section 4
COR002-23-02, p. 67, Table 1

Distractors:

a. Neither RHR loop A pumps are available. The C pump is OOS for maintenance and with 1F bus deenergized no power is available to the A pump.
c. Only RHR pump B still has power availble so only one pump is available. Torus sprays are available from the B loop.
d. No Loop A pumps remain available and torus sprays are not available from RHR loop A.

Source: ***Modified original question 4029.***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	14028	00	04/30/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001080N	Predict the consequences a malfunction of the following will have on the RHR system: Suction flow path
COR0022302001080K	Predict the consequences a malfunction of the following will have on the RHR system: Reactor water level

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
203000.A2.02	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...: (CFR: 41.5 / 45.6) Pump trips (3.5/3.5)

QUESTION: 33 14028 (1 point(s))

"A" loop of RHR is in shutdown cooling at 150°F with "C" RHR pump running.

- ➢ RPV water level drops unexpectedly
- ➢ RPV water level continues to lower to -120 inches Wide Range

3 minutes later, what is the status of "A" and "C" LPCI pumps and what are the MIMIMUM actions are necessary to inject with **BOTH** pumps?
(NOTE: The choices are listed from MIMIMUM to MAXIMUM.)

a. "A" LPCI pump is running, "C" LPCI pump is idle. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START.

b. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START.

c. "A" LPCI pump is running, "C" LPCI pump is idle. Align pump suction paths to the Torus. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

ANSWER: 33 14028

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

<u>Answer source:</u> 2.4SDC p. 13, Attachment 3

Distractors:

a. Both pumps trip and lock out on anti-pump. The SDC isolation must be reset. The suction path must be realigned.
b. The SDC isolation must be reset.
c. Both pumps trip and lock out on anti-pump.

<u>Source:</u> *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	2127	01	03/28/2003	05/21/2003	Electrical	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022602, OPS ROD WORTH MINIMIZER

Related Lessons
COR0022602 OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022602001010A	State the purpose of the following items related to the Rod Worth Minimizer: Rod Worth Minimizer System
COR00226020010300	State the design bases for the RWM system as described in the associated student text.
COR0022602001050M	Briefly describe the following concepts as they apply to the RWM: Minimize clad damage during control rod drop accident (CRDA)

Related References	
4.2	Procedure 4.2, Rod Worth Minimizer

Related Skills (K/A)	
2.1.28	Knowledge of the purpose and function of major system components and controls. (3.2/3.3)

QUESTION: 34 2127 (1 point(s))

Why is the Rod Worth Minimizer required below 10% power but **not** above 10%?

At higher power,

a. fewer rod movements occur which reduces the chances of an error.

b. the Rod Block Monitor prevents fuel damage in the event of a rod drop accident.

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

d. the effects of a rod drop accident are less due to increased moderator temperature causing lower rod worths.

ANSWER: 34 2127

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

Answer source: COR00-2-26-02, p. 9, Section "e"

Distractors:

a. voids reduce rod worths.
b. RBM mitigates rod withdrawal error.
d. Moderator temperature rise increases rod worths.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	14033	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022802, OPS STANDBY GAS TREATMENT

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001100K	Predict the consequences of the following on the Standby Gas Treatment system: Z sump failures

Related References	
2.2.73	Procedure 2.2.73, Standby Gas Treatment System

Related Skills (K/A)	
261000.A1.01	Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: (CFR: 41.5 / 45.5) System flow (2.9/3.1)

QUESTION: 35 14033 (1 point(s))

The plant was operating at power when an LOCA occurred. The reactor building ventilation isolated and SGT automatically started. Additional plant failures occurred that resulted in the failure of Z-sump pumps and a subsequent high level in the Z-sump.

What is the potential effect on SGT?

a. SGT flow decrease

b. Inlet HEPA filter moisture damage

c. Moisture impingment on the SGT fans

d. Increased iodine carryover at the SGT train outlet

ANSWER: 35 14033

a. SGT flow decrease.

The discharge lines have drain lines that are connected to the Z sumps located at the base of the ERP. These SGT discharge lines can become blocked by excessive water level in the Z sump. If water collects in the 10" underground lines, SGT discharge flow may be restricted. Reduced SGT system flow effects the operability of the SGT systems.

Answer source: COR002-28-02, p. 23 & 24, section 3

Distractors:

b. High Z-sump level would not increase the moisture at the inlet of the SGT train.
c. A high level in the Z-sump would not impact the SGT fan.
d. A high level in the Z-sump would not result in increased iodine at the outlet of the train. The reduced flow through the train may even slightly reduce iodine at the outlet.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	14025	00	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022902, STANDBY LIQUID CONTROL

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001120A	Briefly describe the relationships that exist between the SLC system and the following: Core Spray line leak detection

Related References	
COR0022902	SLC
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
211000.K1.09	Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Core spray system: Plant-Specific (3.2*/3.4*)

QUESTION: 36 14025 (1 point(s))

Where do the Core Spray Line Break Detection differential pressure switches (dPIS-43A/B) connect?

The high pressure side of the pressure switch is connected to the SLC sparger to sense pressure . . .

a. below the core plate and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

b. below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

c. above the core plate in the core bypass region and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

ANSWER: 36 14025

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

Downstream of the manual isolation valve (14A/B) each Core Spray system has an instrument line that is connected to the low pressure side of the Core Spray Line Break Detection differential pressure switch. The high pressure side of the dPIS is connected to the Standby Liquid Control "outer" pipe, which detects the pressure in the bypass region above the Core Plate.

<u>Answer source:</u> COR002-29-02 Figure 7

Distractors:

a. The high pressure side senses above the core plate NOT below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.
b. The high pressure side senses above the core plate NOT below the core plate.
c. The low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	5348	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

Related Objectives	
COR0023002001060F	Describe the SRM system design features and/or interlocks that provide for the following: IRM/SRM interlock

Related References	
4.1.1	Procedure 4.1.1, Source Range Monitoring System

Related Skills (K/A)	
215004.K3.02	Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor manual control: Plant-Specific (3.4/3.4)

QUESTION: 37 5348 (1 point(s))

A Reactor Startup is in progress with the following conditions:

- ➢ Power is rising with a stable, positive period
- ➢ SRM detectors are withdrawn except for SRM "A" which fails to withdraw
- ➢ The SRM UPSCALE OR INOPERATIVE alarm has been received
- ➢ The SRM is **NOT** bypassed

As power continues to rise, what is the **FIRST** point that rods will be able to be withdrawn?

a. Associated IRMs are on Range 3 or higher.

b. Associated IRMs are on Range 8 or higher.

c. Associated IRMs are on Range 9 or higher.

d. The Mode switch is placed to RUN.

ANSWER: 37 5348

b. Associated IRMs are on Range 8 or higher.

The SRM Upscale or Inop Rod Block is bypassed when all associated IRM's are selected to range 8 or above.

Answer source: 4.1.1 p. 5, step 1.2.2

Distractors:

a. Range 3 bypasses the detector withdrawal permissive interlock of 100 cps.
c. Range 9 has no bypass functions.
d. While RUN bypasses all SRM Interlocks/Trips, Range 8 will be achieved first.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	19084	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023102, TRAVERSING IN-CORE PROBE

Related Lessons
COR0023102 OPS TRAVERSING IN-CORE PROBE

Related Objectives	
COR0023102001110A	Describe the TIP system design features and/or interlocks that provide for the following: Primary containment isolation
COR0023102001130C	Given a TIP system control manipulation, predict and explain the changes in the following parameters: Valve status
COR0023102001140H	Predict the consequences of the following on the TIP system: High primary containment pressure
COR0023102001160B	Given plant conditions, determine if any of the following TIP actions should occur: Ball valve closure

Related References	
4.1.4	Procedure 4.1.4, Traversing In-Core Probe System

Related Skills (K/A)	
215001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7) Primary containment isolation system: Mark-I&II (Not- BWR1) (3.1/3.4)

QUESTION: 38 19084 (1 point(s))

The plant is at 100% power with TIP traces in progress. Only "A" TIP machine is being used at this time. Currently, TIP "A" has reached the Core Bottom Limit and is moving at slow speed to the Core Top Limit. The IN-CORE light is ON.

A reactor scram due to low RPV water level occurs. One (1) minute later, an operator observes:

- ➢ TIP valve indication on Containment Isolation display (Panel 9-3) is RED
- ➢ **IN-SHIELD** light for TIP "A" is ON at Panel 9-13
- ➢ Drywell pressure is normal

What action is ***required***?

a. Fire TIP "A" shear valve.

b. Close TIP "A" ball valve.

c. Manually retract TIP "A" to fire the shear valve.

d. Manually retract TIP "A" to close the ball valve.

ANSWER: 38 19084

b. Close TIP "A" ball valve.

If red light (Panel 9-3) stays on, at least one TIP ball valve has not closed. After automatic withdrawal of the TIP on the PCIS group 2 isolation signal, the ball valve failed to automatically close. This failed automatic action requires immediate operator action to manually perform the ball valve closure. The procedure directs the operator to attempt to manually retract TIP. Since the TIP is already retracted (IN-SHIELD light is on), this action is not necessary. If ball valve cannot be closed and there are indications of a reactor coolant leak in drywell (as evidenced by the high drywell pressure) then fire appropriate shear valve by operating appropriate keylock switch.

Answer source: 4.1.4 p. 2, step 2.10
4.1.4 p. 8, step 6.3

Distractors:

a. There is no indications of a LOCA and no attempt has yet been made to close the ball valve.
c. The TIP has already retracted and retracting the TIP does not fire the shear valve.
d. The TIP has already retracted.

Source: Direct from Bank

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	14004	00	03/21/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR0023202001070D	Given a RVLC system control manipulation, predict and explain the changes in the following parameters: Controller Indications
COR0023202001060C	Predict the consequences of the following on the RVLC system: Control Signal Failure/Track and Hold

Related References	
2.4RXLVL	RPV WATER LEVEL CONTROL TROUBLE
2.2.28.1	Procedure 2.2.28.1, Feedwater System Operation

Related Skills (K/A)	
259002.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water level control controller indications (3.6/3.6)

QUESTION: 39 14004 (1 point(s))

Given the following conditions:

- Reactor power is 90%
- Reactor water level is 35"
- RVLC is in 3 element
- The Master Controller is in BAL the tape setpoint is 35"
- The selected level instrument **INSTANTANEOUSLY** fails downscale

What is the current configuration of RFC-CS-RFPTA (RFPT A M/A station)?

The RFPT M/A station shifts to ___________ with controller output _____________ controller output prior to the event.

a. Manual Demand Mode (MDEM)
 the same as

b. Manual Direct Valve Positioning Mode (MDVP)
 the same as

c. Manual Demand Mode (MDEM)
 higher than

d. Manual Direct Valve Positioning Mode (MDVP)
 higher than

ANSWER: 39 14004

a. Manual Demand Mode (MDEM)
 the same as

Answer source: New Lovejoy training material (no electronic version). New electronic versions of 2.2.28.1 and 2.4RXLVL not yet available without exam compromise.

Distractors:

b. The controller shifts to MDEM mode.
c. The output won't change.
d. The controller shifts to MDEM mode and the output won't change.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	14014	00	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	COR0023402, Alternate Shutdown (LO)

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR00234020010700	State the design bases for the ASD system as described in the associated Student Text.
COR00234020010100	State the purpose of the Alternate Shutdown system.

Related References	
5.1ASD	Shutdown From Outside The Control Room

Related Skills (K/A)	
295016.AK3.03	Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6) Disabling control room controls. (3.5/3.7*)

QUESTION: 40 14014 (1 point(s))

When the control room is evacuated, why are the ASD panel isolation switches placed in the ISOLATE Position?

a. To prevent spurious equipment operation.

b. To ensure automatic operation of ECCS remains available.

c. To isolate circuits to meet divisional physical separation criteria.

d. To prevent overloading the associated DG during a design basis LOCA.

ANSWER: 40 14014

a. To prevent spurious equipment operation.

Answer source: COR002-34-02 p. 11, section 4

Distractors:

b. Automatic operation of ECCS does NOT remain available.
c. Operation of these switches has nothing to do with divisional separation.
d. Operation of these switches has nothing to do with diesel loading.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	8970	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070501, OPS Introduction to Technical Specifications

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010800	From memory, state each CNS Safety Limit and discuss the basis for each of the Safety Limits.

Related References	
2.1.1.1	With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow: THERMAL POWER shall be < 25% RTP.

Related Skills (K/A)	
290002.K5.07	Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS: (CFR: 41.5 / 45.3) Safety limits (3.9/4.4)

QUESTION: 41 8970 (1 point(s))

The plant was operating at 100% power when a DEH failure results in a reactor pressure reduction. Reactor pressure decreases to 700 psig and reactor power decreases to 65%. The Group 1 isolation fails to actuate. The operating crew scrams the reactor and manually closes the Main Steam Isolation Valves.

What is a potential consequence of this event?

a. Increased likelihood of thermal hydraulic instabilities.

b. The linear heat generation rate limit for some fuel is exceeded.

c. The average planar linear heat generation rate limit is exceeded.

d. The potential is created for radioactive release in excess of 10CFR100 limits.

ANSWER: 41 8970

d. The potential is created for radioactive release in excess of 10CFR100 limits.

The scenario given represents the violation of the fuel integrity safety limit. Reactor power is greater than 25% with reactor pressure less than 785 psig. Exceeding a safety limit may cause fuel damage and create the potential for radioactive releases in excess of 10CFR100.

<u>Answer source:</u> Safety Limit Violation Bases p. B 2.0-5

Distractors:

a. The likelihood of thermal hydraulic instabilities is actually reduced by the decrease in core inlet subcooling casued by the rapid pressure reduction.
b. This reduction in power is global resulting in a power reduction for each core bundle. This would increase the margin to LHGR limit.
c. With the global reduction in power average linear heat generation rate would decrease and put operation farther away from the limit.

<u>Source:</u> *Modified from 8970*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	3995	01	12/31/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	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	
INT0070502001060A	From memory, for MODES 1 and 2, state the actions required in less than one hour for: two or more control rod scram accumulators inoperable with the reactor steam dome pressure less than or equal to 940 psig (LCO 3.1.5 B.1).

Related References	
3.1.5	Control rod scram accumulators

Related Skills (K/A)	
295022.AK2.03	Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: (CFR: 41.7 / 45.8) Accumulator pressures. (3.4/3.4)

QUESTION: 42 3995 (1 point(s))

Given the following conditions:

- A Reactor startup **AND** heatup is in progress
- Reactor power is 3%
- Reactor Steam Dome Pressure is 835 psig
- Control Rod Drive Hydraulic Pump 1A trips and will not restart
- Control Rod Drive Hydraulic Pump 1B will not start

What action(s) are required by Technical Specifications for these conditions?

When the accumulator pressure is < 935 psig for . . .

a. **ANY** control rod, immediately declare the associated Control Rod inoperable.

b. **ANY** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

d. one **withdrawn** control rod, restore Charging Header pressure to greater than 940 psig within 20 minutes.

ANSWER: 42 3995

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

Tech Spec 3.1.5 Condition C applies. With RPV Pressure < 900 psig, withdrawn rods with inoperable accumulators may fail to scram under low pressure conditions and must be immediately inserted. Since the rod cannot be inserted without drive pressure, Condition D applies and the Reactor must be scrammed.

Answer source: Actions per LCO 3.1.5 for reactor pressure < 900 psig (RA C.1 & D.1).

Distractors:

a. This is the action for a slow control rod with an inoperable accumulator.
b. The control rod must be withdrawn to require the scram.
d. This is the action if Reactor Pressure is > 900 psig.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	14048	00	03/28/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	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	
INT00705040010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Related References	
3.3.2.2	Feedwater and main turbine high water level trip instrumentation

Related Skills (K/A)	
295014.AK3.01	Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.5 / 45.6) Reactor SCRAM. (4.1*/4.1)

QUESTION: 43 14048 (1 point(s))

Why does Technical Specifications require the main turbine to trip on high reactor water level?

a. To indirectly prevent damage to the Moisture Separators by low enthalpy fluid during a failure of selected RPV water level instrument.

b. To prevent ECCS equipment damage from missiles created by main turbine failure during a feedwater controller maximum demand failure.

c. To ensure flow induced vibration of the main steam lines remains within analytical limits during a failure of selected RPV water level instrument.

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

ANSWER: 43 14048

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

Per 3.3.2.2 bases "The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event. The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR."

Answer source: Tech Spec Bases p. 3.3-55, Safety Analysis

Distractors:

a. This is not the basis.
b. The missile damage potential is not assumed by the accident analysis.
c. There is no flow induced vibration related to the main turbine trip, but is related to Recirculation flow mismatch.

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	14042	00	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

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

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.
INT00705050010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.4 LCO, determine the ACTIONS that are required.

Related References	
6.LOG.601	Daily Surveillance Log (Tech Specs)

Related Skills (K/A)	
268000.K3.04	Knowledge of the effect that a loss or malfunction of the RADWASTE will have on following: (CFR: 41.5 / 45.3) Drain sumps (2.7/2.8)

QUESTION: 44 14042 (1 point(s))

The plant was operating at power when the Sump F Totalizer (RW-FQ-527) failed.

What is required?

Place/Leave . . .

a. **either** F-1 or F-2 sump pumps in AUTO and repair the totalizer within 4 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

b. **both** F-1 and F-2 sump pumps in AUTO and repair the totalizer within 12 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

c. **either** F-1 or F-2 sump pump switches in PULL-TO-LOCK and leave the other pump in AUTO. Record each time the pump in AUTO pumps.

d. **both** F-1 and F-2 sump pump switches to PULL-TO-LOCK. At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation.

ANSWER: 44 14042

d. F-1 and F-2 sump pump switches to PULL-TO-LOCK.

At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation. When the Sump F Totalizer failed, 6.LOG.601 requires that the total gallons be calculated per Sump F Totalizer table. This table directs that both sump pump switches be placed in Pull-To-Lock and on 8 hour intervals and also when Sump F high alarm is received, pump sump using one pump and time seconds of operation.

Answer source: 6.LOG.601 pp. 9 (Note a) & 10 (DETERMINATION OF TOTAL GALLONS WITH FAILED SUMP F TOTALIZER)

Distractors:

a. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
b. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
c. Neither pump is left in AUTO.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	6226	03	05/16/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320128, CNS Abnormal Procedures (RO) Containment

Related Lessons	
INT0320128	CNS Abnormal Procedures (RO) Containment

Related Objectives	
INT0320128J0J0100	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	
2.3_9-3-2	PANEL 9-3 - ANNUNCIATOR 9-3-2
2.4PC	PRIMARY CONTAINMENT CONTROL

Related Skills (K/A)	
295010.AK1.03	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) Temperature increases. (3.2/3.4)

QUESTION: 45 6226 (1 point(s))

The plant is operating at 100% power when the following conditions exist:

- ➢ Annunciator H-1/A-2 DRYWELL ZONE 1 HIGH TEMP is alarming
- ➢ Annunciator H-1/A-1, DRYWELL FCU A HIGH DISCH TEMP is alarming
- ➢ Annunciator 9-5-2/F-3, DRYWELL HIGH PRESSURE is alarming
- ➢ Two (2) of the Drywell FCU's have tripped

What actions are required of the operator as identified in Abnormal Procedure 2.4PC, PRIMARY CONTAINMENT CONTROL?

a. Maintain drywell pressure ≤ 0.75 psig by venting via SGT, **ONLY**.

b. Start tripped drywell FCUs by placing their control switches to OVERRIDE, **ONLY**.

c. Maintain drywell pressure ≤ 0.75 psig by venting via SGT **AND** if drywell pressure cannot be maintained below 1.5 psig, scram the Reactor.

d. Start tripped drywell FCUs by placing their control switches to OVERRIDE **AND** maintain drywell pressure ≤ 0.75 psig by venting via SGT.

ANSWER: 45 6226

c. Maintain drywell pressure ≤ 0.75 psig by venting via SGT **AND** if drywell pressure cannot be maintained below 1.5 psig, scram the Reactor.

<u>Answer source:</u> PR 2.4PC

Distractors:

a. The ONLY does not include the required scram.
b. Cannot place FCUs in OVERRIDE until in EOP-3A and does not include required scram.
d. Incorrect as the does not include the required scram and cannot place FCUs in OVERRIDE until in EOP-3A.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	5247	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Related References	
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 46 5247 (1 point(s))

The plant is at 35% power when Turbine vibration required the operator to insert a manual scram and trip the turbine. The following conditions exist one (1) minute after the operator has depressed both manual reactor scram pushbuttons:

- ➢ Turbine Bypass Valves are 75% open **(stable)**
- ➢ Main Steam Isolation Valves (MSIVs) are open
- ➢ RPV Narrow Range level is 25" **AND steady**
- ➢ RPV pressure is 940 psig **AND steady**

Which procedure(s) must be executed?

2.1.5 "Reactor Scram" . . .

a. **ONLY**.

b. **AND** EOP-1A "RPV Control" **ONLY**.

c. **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)" **ONLY**.

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

ANSWER: 46 5247

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

Bypass valves at 75% open is approximately 19% power. 19% power after a scram is an entry condition to 1A. EOP 1A directs 6A and 7A to be entered.

Answer source: EOP-1A, 2.1.5 p. 1

Distractors:

a. EOP 1A must be entered.
b. Entry into 6A & 7A is required.
c. Entry into 1A is required.

Source: *Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	13407	0	04/12/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050010900	Identify any EOP support procedures addressed in Flowchart 1A and apply any associated special operating instructions or cautions.

Related References	
5.8.2	Procedure 5.8.2, Alternate Emergency Depressurization Systems (Table 2)

Related Skills (K/A)	
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10) (2.9/3.3)

QUESTION: 47 13407 (1 point(s))

While responding to a LOCA, the RCIC overspeed trip must be reset to allow it to be used as an injection system.

- ➢ The TSC is NOT operational.
- ➢ Several Reactor Building ARMs that an operator must pass within 10 feet of to get to the area are alarming and indicate upscale.
- ➢ Both high range Drywell radiation monitors read 1.5E4 R/hr.

In addition to standard RP practices, what additional (if any) <u>MINIMUM</u> requirement must be met to dispatch an operator to perform this task?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. No additional requirements must be met.

b. A survey instrument that monitors radiation dose rates must be taken.

c. A RP Technician must accompany the operator.

d. The TSC must be declared operational.

ANSWER: 47 13407

d. The TSC must be declared operational.

If DRYWELL RAD MONITOR RMA-RM-40A or DRYWELL RAD MONITOR RMA-RM-40B (Panel 9-02) is reading $\geq$ 1E4 rem/hour, entry into Secondary Containment is prohibited until TSC is operational and personnel can be dispatched per Procedure 5.7.15. The operator cannot be dispatched until the TSC is operational because both drywell radiation monitors are above 1E4 rem/hr.

<u>Answer source:</u> 5.8.2 p. 12, section 5.1

Distractors:

a. The TSC must also be operational to dispatch the operator.
b. The TSC must also be operational to dispatch the operator.
c. The TSC must also be operational to dispatch the operator.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	14000	00	02/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
COR0020302	CONTAINMENT
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
COR0020302001210B	Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations.
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 48 14000 (1 point(s))

RPV water level lowers to 20 inches below the value that is an entry condition for EOP-1A, "RPV Control". No other EOP entry conditions are satisfied.

What group isolations have automatically occurred?

Group 2 . . .

a. **only**.

b. and Group 3 **only**.

c. and Group 3 and Group 6 **only**.

d. and Group 1 and Group 3 and Group 6.

ANSWER: 48 14000

c. and Group 3 and Group 6 **only**.

Answer source: 2.1.22 p. 7, section 5.3
2.1.22 p. 10, section 6.3
2.1.22 p. 17, section 9.6

Distractors:

a. Group 3 and Group 6 also occur.
b. group 6 also occurs.
d. Group 1 does not occur.

2.4.2 Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) as it applies to: 295009 Low Reactor Water Level

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	2403	01	05/19/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020902, Digital Electro-Hydraulic Control

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040C	Describe how the DEH control system operates to control the following: Reactor/turbine pressure regulating system load set/reference
COR0020902001040G	Describe how the DEH control system operates to control the following: Reactor steam flow
COR0020902001040H	Describe how the DEH control system operates to control the following: Main Turbine steam flow

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

Related Skills (K/A)	
295025.EA1.02	Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6) Reactor/turbine pressure regulating system (3.8/3.8)

QUESTION: 49 2403 (1 point(s))

With DEH load limit set at 300 MW, reactor power is raised from 250 to 400 MWE.

What will be the final Steam Flow and Main Turbine Bypass Valve (BPV) conditions?

a. Rx. steam flow = turbine steam flow, BPVs closed.

b. Rx. steam flow < turbine steam flow, BPVs closed.

c. Rx. steam flow > turbine steam flow, BPVs partially open.

d. Rx. steam flow = turbine steam flow, BPVs partially open.

ANSWER: 49 2403

c. Rx. steam flow > turbine steam flow, BPVs partially open.

As power increases, reactor pressure will rise. The rise in reactor pressure will raise equalizing header pressure, causing governor valves to open until load limit is met. Then, the increase in pressure will cause the Main Turbine Bypass Valves to open to maintain pressure.

Answer source: COR002-09-02, p. 9, section "c."

Distractors:

a. Turbine steam flow is limited to 300 MW, BPVs will be open.
b. Rx steam flow will be > turbine steam flow, BPVs will be open.
d. Turbine steam flow is limited to 300 MW.

Source: *Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	19034	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080618	EOP AND SAG GRAPHS
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.
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.
INT00806050011200	Given plant conditions, assess if RPV water level can be determined or not.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
226001.A1.02	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: (CFR: 41.5 / 45.5) Containment/drywell temperature (3.4/3.5)

QUESTION: 50 19034 (1 point(s))

A Loss of Coolant Accident has occurred with the following conditions:

➢ Reactor pressure	270 psig (lowering at 5 psig/minute)
➢ Indicated water level	-120" (Wide range, steady)
➢ Drywell pressure	5.5 psig (rising slowly)
➢ Drywell temperature	350° F (all points) (steady)

What is the status of Wide Range Reactor Level Instrumentation ***now*** and what effect (if any) will drywell sprays have on ***future*** availability?

Wide Range Reactor Level Instrumentation . . .

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

b. can be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

c. **CANNOT** be used for trending.
Initiation of drywell sprays will restore instrument availability.

d. **CANNOT** be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

ANSWER: 50 19034

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes. Initiation of DW sprays cools containment and prevents intrusion into the unsafe region of reactor saturation graph (EOP Graph 1), in addition DW sprays and their cooling effect on DW temperature ensure that the instrumet **stays** in the safe region of Graph 15.

Answer source: EOP Graph 1 & EOP Graph 15, effects of drywell spray

Distractors:

b. Drywell sprays will maintain Instrument availability.
c. The instruments are currently available for trending.
d. The instruments are currently available for trending and drywell sprays will maintain Instrument availability.

<u>Source:</u> ***Direct from bank***

<u>Provide to Candidate:</u> EOP Graphs

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	14001	00	02/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010600	List the methods of alternate rod insertion.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295015.AK2.04	Knowledge of the interrelations between INCOMPLETE SCRAM and the following: (CFR: 41.7 / 45.8) RPS (4.0/4.1)

QUESTION: 51 14001 (1 point(s))

The plant was operating at 100% power with a half scram on RPS A due to a relay failure and 1B CRD pump tagged out of service. Subsequently, power was lost to RPSPP 1B. The following conditions now exist:

- Many control rods failed to insert
- Reactor power is 5%
- 4160 VAC Bus 1F is de-energized

What method can the crew use to successfully insert the control rods that failed to insert on the scram?

a. Individually scram control rods.

b. Drain the SDV and manually scram.

c. Manually vent CRDM over-piston area.

d. Insert control rods using emergency override switch.

ANSWER: 51 14001

c. Manually vent CRDM over-piston area.

The multiple RPS failures result in the inability to reset the scram. In addition, having the 1B CRD pump out of service in conjunction with the loss of power to the operating CRD pump precludes the use of RMCS to drive control rods. Venting the over piston area of the CRDM is the only action listed that is consistent with 5.8.3, Alternate Rod Insertion Methods.

Answer source: 5.8.3 flowchart, path "B"

Distractors:

a. The scram valves cannot be closed due to the RPS failures.
b. Draining the SDV and manually scramming would require that the scram valves be closed which is not possible with the RPS failures.
d. RMCS cannot be used to drive control rods because there are no operating CRD pumps.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	16472	01	03/24/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060010600	List the methods of alternate rod insertion.
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010900	Identify any EOP support procedures addressed in Flowchart 6A and apply any associated special operating instructions or cautions.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295037.EK2.05	Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: (CFR: 41.7 / 45.8) CRD hydraulic system (4.0/4.1)

QUESTION: 52 16472 (1 point(s))

The plant scrammed from full power, but many control rods failed to insert (see following figure). Control rods will be inserted using the Reactor Manual Control System.

Which one of the following rod insertion sequences should be used?
(NOTE: Blank positions on the figure indicate control rod is fully inserted.)

Fully insert rod . . .

a. 26-27, then fully insert 26-39, then 30-27, then 34-27 and continue to insert rods that have high worth.

b. 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

c. 30-27, then fully insert 18-27, then 30-35 and continue to criss-cross the core outward from the center of the core.

d. 30-27, then fully insert 22-27, then 22-23 and continue this spiral pattern inserting the most withdrawn control rods first.

	02	06	10	14	18	22	26	30	34	38	42	46	50
51													
47													
43													
39				10			32	36					
35					12	26		36					
31				12		36			10				
27					12	48	46	42	32				
23						48			24				
19					10		34	12					
15													
11													
07													
03													

ANSWER: 52 16472

b. Fully insert rod 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

Under ATWS conditions, control rod insertions should begin in the center of the core, and proceed to every other control rod in an outward spiral pattern.

Answer source: 5.8.3 p. 4, section 4.2

Distractors:

a. Skips control rods, this method has no real pattern to it.
c. Does NOT start with the center rod and does not insert every other rod by geometric location.
d. Does NOT start with the center rod and does not insert every other rod by geometric location.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
53	16485	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	6	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM

Related Lessons	
INT0080610	OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM
INT0080612	FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM
INT0080602	OPS FLOWCHART ORGANIZATION AND STRUCTURE

Related Objectives	
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.
INT00806120010200	Describe the conditions required to assure adequate core cooling while flooding the core during a failure to scram transient.
INT00806020010800	List the three mechanisms used in the EOP flowcharts to assure adequate core cooling.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295015.AK1.04	Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10) Reactor pressure: Plant-Specific. (3.8/3.8)

QUESTION: 53 16485 (1 point(s))

Which one of the following conditions, by itself, assures fuel clad temperature does not exceed 1500°F ***<u>during an ATWS</u>***?
(Note: All RPV levels are as INDICATED on the Fuel Zone instruments.)

Reactor Pressure . . .

a. 197 psig, ***indicated*** RPV level -50 inches, Two (2) SRVs open.

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

c. 452 psig, ***indicated*** RPV level -66 inches, One (1) SRV open.

d. 790 psig, ***indicated*** RPV level -60 inches, **NO** SRVs open.

ANSWER: 53 16485

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

Actual level in all 4 choices is below -25 inches for adequate core cooling with level/steam cooling. Minimum Steam Cooling Pressure met only in choice "b" (MSCP for 3 SRVs is 280 psig).

<u>Answer source:</u> EOP flowchart 7A steps FS/L-12 & FS/L-17
EOP Flowcharts 7A & 7B, Table 14
INT008-06-06 p.20

Distractors:

a. Below -25" actual, below MSCP.
b. Below -25" actual, below MSCP.
d. Below -25" actual, below MSCP.

<u>Source:</u> ***Modified***

<u>Provide to Candidate:</u> EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	54	0	05/19/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM

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

Related Objectives	
INT00806100010700	Identify any EOP support procedure addressed in Flowchart 7A and apply any associated special operating instructions or cautions.
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References
NONE

Related Skills (K/A)	
295031.EA1.12	Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) Feedwater. (3.9/4.1*)

QUESTION: 54 54 (1 point(s))

A Manual reactor scram was inserted from rated power. The following conditions exist:

- Reactor power is 30%
- Reactor water level is 35"
- The Master Controller is in BAL the tape setpoint is 35"

What is the correct sequence of actions to Stop and Prevent injection from the Reactor Feedwater pumps?

a. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Demand Mode (MDEM) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

b. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Direct Valve Positioning Mode (MDVP) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

c. Close the Reactor Feedwater pump discharge valves and trip the Reactor Feedwater pumps. Then ensure RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) are in Manual Demand Mode (MDEM) with output at minimum.

d. Close the Reactor Feedwater pump discharge valves and trip the Reactor Feedwater pumps. Then ensure RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) are in Manual Direct Valve Positioning Mode (MDVP) with output at minimum.

ANSWER: 54 54

b. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Direct Valve Positioning Mode (MDVP) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

The feedwater pump is placed in MDVP mode and reduced to minimum before closing the discharge valves and tripping the reactor feedwater pumps. This sequence of events rapidly reduces feedwater flow.

<u>Answer source:</u> Procedure 5.8 page 4.

Distractors:

a. The controller is placed in MDVP mode.
c. The RFP is tripped after the controller is adjusted. The controller is placed in MDVP mode.
d. The RFP is tripped after the controller is adjusted. The controller is placed in MDVP mode.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	14047	00	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295013.AA2.01	Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.8/4.0)

QUESTION: 55 14047 (1 point(s))

Given the following conditions:

- ➢ Primary Containment water level 14 feet
- ➢ Reactor pressure 800 to 1000 psig (with SRVs)

Which of the following is the **LOWEST** average suppression pool water temperature which requires Emergency Depressurization?

a. 195°F

b. 205°F

c. 211°F

d. 215°F

ANSWER: 55 14047

b. 205°F

Answer source: EOP graph 7
EOP flowchart 3A step SP/T-5

Distractors:

a. Too low.
c. Too high.
d. Too high.

Source: ***New***

Provide to Candidate: EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	5268	02	02/26/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295012.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Drywell pressure (3.9/4.1)

QUESTION: 56 5268 (1 point(s))

Given the following conditions:

- A small coolant leak has occurred in the Drywell
- Drywell temperature is 185°F and rising slowly
- Drywell pressure is 4.0 psig and rising slowly

What action is required to control Drywell conditions?

a. Initiate drywell sprays.

b. Operate all available drywell cooling.

c. Vent primary containment with torus vent line.

d. Vent primary containment with drywell vent line.

ANSWER: 56 5268

b. Operate all available Drywell Cooling.

This action is specified in the drywell temperature leg of EOP-3A when drywell temperature cannot be maintained below 150°F.

Answer source: EOP flowchart 3A step DW/T-3

Distractors:

a. Drywell sprays are not permitted with torus pressure at the current 2.6 psig.
c. Venting the torus is not allowed with the LOCA signal present and pressure well below PCPL-A.
d. Venting the drywell is not allowed with the LOCA signal present and pressure well below PCPL-A.

Source: *Modified 5268*

Provide to Candidate: EOP-3A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	5332	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS 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.
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.1.20	Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) (4.3/4.2)

QUESTION: 57 5332 (1 point(s))

Given the following conditions:

- A Loss of Coolant Accident has occurred
- Reactor pressure is 590 psig
- Drywell pressure is 6.5 psig **AND** rising
- Drywell temperature is 200° F
- Drywell Spray is in service with suction ***from the torus***
- Torus Temperature is 165° F
- Torus level is 16.8 feet **AND** rising
- Torus spray is in service

What containment spray action is required **AND** what is the bases for the action?

a. Terminate Drywell Spray to stop the water addition to the Torus.

b. Terminate Torus Spray since the Torus Spray Header is submerged.

c. Terminate Torus Spray to raise Torus pressure to drive non-condensable gases into the Drywell.

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

ANSWER: 57 5332

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

<u>**Answer source:**</u> EOP flowchart 3A steps DS-1 & DS-2
INT008-06-13 p. 11

Distractors:

a. Drywell Spray water can be taken from the Torus.
b. The Spray header is submerged at 26.75'.
c. The Vacuum Breakers will not pass sufficient flow when covered.

2.1.20 Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) as it applies to: 295029 High Suppression Pool Water level / 5

<u>Source:</u> ***Direct from bank***

<u>Provide to Candidate:</u> EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	5333	00	11/16/1999	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
500000.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6) Emergency depressurization (3.1/3.9)

QUESTION: 58 5333 (1 point(s))

What hydrogen **AND** oxygen concentration values require Emergency Depressurization **AND** what is the bases for the Emergency Depressurization?

Containment hydrogen AND oxygen concentrations require an Emergency Depressurization at _________________ in order to ensure the . . .

a. 5% H_2 **AND** 6% O_2
source of hydrogen production is removed.

b. 6% H_2 **AND** 5% O_2
source of hydrogen production is removed.

c. 5% H_2 **AND** 6% O_2
Reactor is at the lowest energy state possible.

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

ANSWER: 58 5333

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

Answer source: EOP flowchart 3A Table 7
INT008-06-13 p. 18

Distractors:

a,b. Combustible limits are 6% hydrogen and 5% oxygen. Depressurization does not remove the hydrogen source.
c. Concentrations are reversed.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	14023	00	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295028.EK3.05	Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.5 / 45.6) Reactor SCRAM (3.6/3.7)

QUESTION: 59 14023 (1 point(s))

When performing actions in EOP-3A "Primary Containment Control" for high drywell temperature, what is the reason for entering **AND** performing EOP-1A "RPV Control" concurrently without a specific EOP-1A entry condition being met?

Entering EOP-1A "RPV Control" and inserting a manual scram ensures . . .

a. the power produced by the reactor will be within the Primary Containment vent capability.

b. the RPV is at the lowest possible energy state before implementing more severe strategies.

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

d. the main source of potential energy addition to the Primary Containment is removed before conditions warrant Emergency Depressurization.

ANSWER: 59 14023

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

Answer source: INT008-06-13 p. 22

Distractors:

a. RCIC exhaust flowrate, factors for basis of PCPL-A.
b. The "lowest energy state" is an Emergency Depressurization basis. There is no more severe strategy than Emergency Depressurization.
d. It is not a "potential" energy addition and scramming the reactor does not remove the source of the energy addition.

Source: *New*

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	19082	00	05/16/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure
5.9H2O2	Severe Accident Procedure (Primary Containment Combustible Gas Control [SAG 3])

Related Skills (K/A)	
223001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: (CFR: 41.7 / 45.7) Combustible gas mixing: Plant-Specific (2.8/2.8)

QUESTION: 60 19082 (1 point(s))

A LOCA has occurred and the following conditions exist:

- Drywell H_2 9% (rising slowly)
- Drywell O_2 4% (steady)
- Suppression Chamber H_2 4% (rising slowly)
- Suppression Chamber O_2 6%. (steady)
- Drywell Spray Unavailable

What effect does the lack of Drywell sprays have on the Primary Containment and what action is required?

The inability to initiate drywell sprays . . .

a. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (within ODAM release rate limits) is required.

b. decreases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (within ODAM release rate limits) is required.

c. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

d. decreases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

ANSWER: 60 19082

c. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

The lack of drywell sprays prevents mixing of combustible gasses and increases gas concentrations. The lack of sprays into the atmosphere also increases flamability. These indications are sufficient for Combustiblity. SAP 5.9H2O2 requires emergency venting the PC irrespective of off-site release rates.

Answer source: EOP-3A; PC/G-2 and TABLE 7.
SAP5.9H202

Distractors:

a. Required to vent irrespective of release rate limit.
b. Increases chance of deflagration, required to vent irrespective of release rate limit.
d. Increases chance of deflagration.

<u>Source:</u> ***New***

<u>Provide to Candidate:</u> EOP-3A with cautions and entry conditions removed. 5.9H2O2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	5730	03	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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

Related Objectives	
INT00806170010400	State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 61 5730 (1 point(s))

What is the basis for the value selected as the EOP Flowchart 5A Maximum Safe Operating (MSO) Radiation level?

It is based on personnel exposures exceeding the . . .

a. CNS TEDE dose limit of 3 Rem ***per quarter*** for a **one hour** stay time.

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

c. 10CFR20 planned special exposure limit of 3 rem for a **one hour** stay time.

d. 10CFR20 planned special exposure limit of 5 rem for a **two hour** stay time.

ANSWER: 61 5730

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

Answer source: INT008-06-17 pp. 10 & 11

The MSO radiation level of 1000 mr/hr (Table 10) is based on personnel exposure assuming:

- The CNS TEDE dose limit of 4 Rem (4000 mrem) per year (Procedure 9.ALARA.1) derated by 0.67 (1.5 safety factor).

- A two hour stay time. (Two hours is the maximum time an individual would be expected to remain in secondary containment during emergency conditions.)

- An assumed internal dose contribution of 20% of the external dose contribution (threshold for respiratory protection from an ALARA standpoint, Procedure 9.ALARA.5).

Or, 4000 mrem/(1.2*2*1.5 hours) = 1111 mrem /hour. A Maximum Safe Opeating radiation level of 1000 mrem/hour is specified because it is reasonably close to the 1111 mrem/hour value and much easier to remember and read. Health Physics access control determines the method of monitoring for this limit.

Distractors:

a. The quarterly limit no longer exists, but it was previously the basis for MSO radiation level, and herefore remains a plausible distractor.

c, d. MSO Radiation levels are not based on the 10CFR20 limits for planned special exposures.

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements as it applies to 290001 Secondary Containment

Source: ***Modified from 5730***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	14019	00	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295038.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) Emergency depressurization (3.6/3.9)

QUESTION: 62 14019 (1 point(s))

What is the basis for performing an Emergency Depressurization during the execution of EOP 5A, **RADIOACTIVITY RELEASE CONTROL**?

Performing an Emergency Depressurization ensures the . . .

a. availability of equipment in the turbine building that may be necessary to mitigate the event is not challenged.

b. energy level of the radiation and the atomospheric dispersion factors fall within the bounds of the accident analysis.

c. isotopic mixture of radioactive materials deposited off-site will be within the bounds of the accident analysis.

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

ANSWER: 62 14019

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

Answer source: INT008-06-17 p. 14

Distractors:

a. availability of turbine building equipment is not an EOP consideration, but sounds like the reactor building ED reason.
b. This is a reason to use the DOSE program for the projections vice the ODAM calculations, but is not the reason for the ED.
c. This is not the basis for the ED.

Source: ***New***

Provide to Candidate: EOP 5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	14021	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295036.EK3.02	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.5 / 45.6) Reactor SCRAM. (2.8/2.8)

QUESTION: 63 14021 (1 point(s))

Why is a reactor scram required before exceeding a Maximum Safe Operating Water Level if a primary system is discharging into the Reactor building (EOP-5A step SC-12)?

Scramming the Reactor . . .

a. provides mitigating action such that the condition does not pose an immediate threat to the health and safety of the public.

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

c. promptly reduces to decay heat levels the energy discharged into primary containment and reduces the driving head and flow of primary systems that are unisolated.

d. promptly places the primary system in its lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.

ANSWER: 63 14021

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

Answer source: INT008-06-17 p. 10, section 10

Distractors:

a. Not a bases for scramming.
c. Primary Containment Control bases.
d. Bases for emergency depressurization.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	16483	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

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

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

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

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 64 16483 (1 point(s))

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- ➢ RCIC cannot be isolated
- ➢ RCIC temperatures in the NE Quad are 214°F and rising
- ➢ Reactor Building ventilation exhaust radiation levels are 25 mr/hr
- ➢ RCIC Room radiation levels are 300 mr/hr and rising
- ➢ A reactor scram is inserted and all control rods fully insert

Five (5) minutes later, annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD clears.

What is the required EOP response to this annunciator clearing and what is the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Do **NOT** restart Reactor Bldg. HVAC because EOP 1A requires a group 6 isolation.

d. Do **NOT** restart Reactor Bldg. HVAC until RP ensures normal radiation levels to minimize the spread of contamination.

ANSWER: 64 16483

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

Answer source: INT008-06-17 p. 7, section "b"

Distractors:

a. This is not a bases for restarting Reactor Bldg. HVAC
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	19068	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.11	Ability to control radiation releases. (CFR: 45.9 / 45.10) (2.7/3.2)

QUESTION: 65 19068 (1 point(s))

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: 65 19068

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

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.

Answer source: INT008-06-17 p. 13, 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.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	14046	00	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080618, EOP AND SAG GRAPHS

Related Lessons
INT0080618 EOP AND SAG GRAPHS

Related Objectives	
INT00806180010100	Using the graphs provided in the EOP and SAG Graphs Flowchart, determine how the shape of each curve or family of curves was determined.

Related References	
PSTG	Plant Specific Technical Guideline
INT0080618	EOP and SAG Graphs and Cautions

Related Skills (K/A)	
295030.EK1.03	Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) Heat capacity. (3.8/4.1*)

QUESTION: 66 14046 (1 point(s))

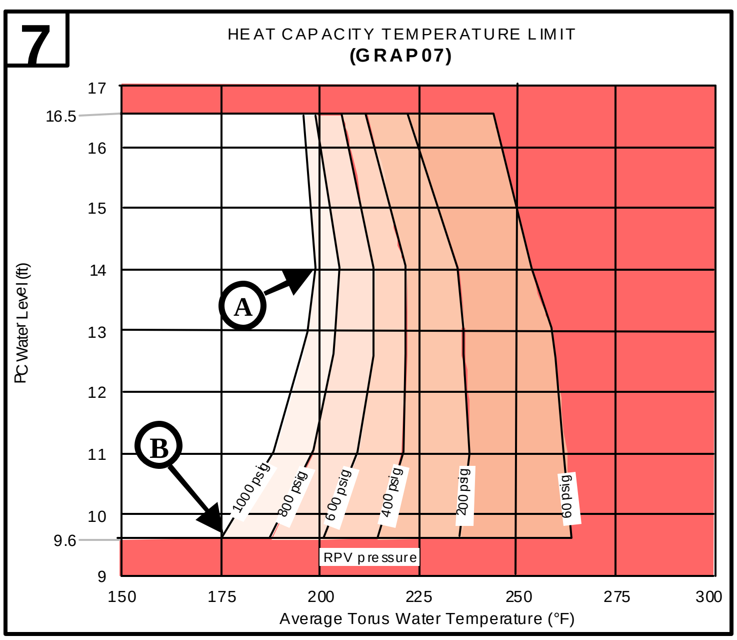
What is the bases for the shape of the Heat Capacity Temperature Limit (GRAPH 07) from point "A" to point "B"?

a. As torus level lowers, there is less backpressure on the SRV tailpipes. With lower SRV backpressure, a lower initial temperature will exceed analyzed blowdown rates during a LOCA.

b. As torus level lowers, there is less backpressure on the downcomers. With lower downcomer backpressure, a lower initial temperature will exceed peak design containment pressure during a LOCA blowdown.

c. Torus airspace volume increase has a greater effect on the limit than torus heat sink mass decrease. As torus airspace volume rises, a lower temperature is needed to heat and pressurize the airspace to PCPL-A.

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.



ANSWER: 66 14046

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.

Answer source: INT008-06-18 pp. 12 & 13

Distractors:

a. HCTL is based on not exceeding PCPL-A and has nothing to do with SRV backpressure.
b. HCTL is based on not exceeding PCPL-A and has nothing to do with downcomer backpressure.
d. Torus airspace volume increase as a lesser effect and acts to make a less restrictive limit rather than a more resrictive limit.

Source: ***New***

Provide to Candidate: EOP Graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	4243	01	01/26/2000	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)

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

Related Objectives	
INT032010100G010M	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Frequency

Related References	
0.26	Surveillance Program

Related Skills (K/A)	
2.2.12	Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) (3.0/3.4)

QUESTION: 67 4243 (1 point(s))

A monthly surveillance test was completed at 1300 on May 4th (SR 3.6.4.3.1, Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.).

In accordance with procedure 0.26 (Surveillance Program), what is the **LATEST** this test can be scheduled to be completed, without exceeding an LCO?

a. 1300 on June 4th

b. 1300 on June 5th

c. 1300 on June 11th

d. 0700 on June 12th

ANSWER: 67 4243

d. 0700 on June 12th

A deviation of 25% of the monthly surveillance interval (38.75) is allowed

Answer source: 0.26 Surveillance Program Page 10 Section 8.1.5

Distractors:

a. Using 30 day as montly with no extension.
b. Using 31 day as montly with no extension.
c. Using 38 vice 38.75.

2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) as it applies to: 261000 SGTS

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	12215	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320102, CNS Administrative Procedures Site Services Procedures (Form

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure
INT032010200C010B	Discuss the following as described in Administrative Procedure 1.15, Visitor/Tour Station Access: Vital Area access criteria

Related References	
1.15	Visitor/Tour Station Access

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 68 12215 (1 point(s))

Four vendor representatives were provided an escorted tour to perform a walkdown of the RHR system to assist in correcting an emergent problem. TLDs were not issued to the vendors. Following the walkdown, the following dose was received by each of the vendors on their DRDs:

- Rick Cox 106 mrem
- Ira Fox 98 mrem
- Marvin Estes 51 mrem
- Robert Smith 101 mrem

Which of the following are required?

An Exposure History Worksheet must be completed for . . .

a. **<u>all</u>** the visitors prior to departure.

b. **<u>only</u>** Rick Cox before departure.

c. **<u>only</u>** Rick Cox and Robert Smith before departure.

d. **<u>only</u>** Rick Cox, Robert Smith and Ira Fox before departure.

ANSWER: 68 12215

c. **<u>only</u>** Rick Cox and Robert Smith before departure.

If the dose received is > 100 mrem (1.0 mSv) by DRD, or equivalent, and no TLD was issued, Radiological Protection shall be contacted and a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing the visitor to depart. Both these individuals received greater than 100 mrem and since TLDs were not issued, a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing they depart.

<u>Answer source:</u> 1.15 page 4, step 4.6

Distractors:

a. Ira Fox and Marvin Estes received less than 100 mrem and are not required to complete CNS-RP-10, Exposure History Worksheet.
b. Robert Smith is also required to complete CNS-RP-10, Exposure History Worksheet prior to departure.

d. Ira Fox received less than 100 mrem and is not required to complete CNS-RP-10, Exposure History Worksheet.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	12209	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320103, CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300E010B	Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Alarm acknowledgement

Related References	
2.3.1	Procedure 2.3.1, General Alarm Procedure
OI-07	Operations Management Expectations

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 69 12209 (1 point(s))

The plant is operating at 100% power. Surveillance Procedure 6.2PCIS.302, MAIN STEAM LINE HIGH TEMPERATURE CHANNEL FUNCTIONAL TEST 9 (DIV 2) has just commenced.

STEAM TUNNEL HIGH TEMP CHANNEL A (9-5-1/E-1) then alarms (the first annunciation of this alarm for the shift).

What action(s) is/are required?

Acknowledge the annunciator . . .

a. **only**.

b. **AND** announce "expected annunciator".

c. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **only**.

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

ANSWER: 69 12209

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

This annunciator is not associated with the surveillance and therefore the alarm should be acknowledged, announced and the alarm card pulled.

<u>Answer source:</u> 2.3.1 p. 3, steps 4.12, 4.13 & 4.14

Distractors:

a. This annunciator is not expected for this surveillance.
b. This annunciator is not expected for this surveillance.
c. The alarm card must be pulled for this annunciator.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	13673	01	06/18/2002	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Procedures	INT0320104, CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

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

Related Objectives	
INT032010400G0300	State from memory the " Mitigating Task Scram Actions" associated with Procedure 2.1.5, Reactor Scram.

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
295006.AA2.05	Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) Whether a reactor SCRAM has occurred (4.6*/4.6*)

QUESTION: 70 13673 (1 point(s))

The plant was operating at full power when a reactor scram occurred.

Per 2.1.5, "Reactor Scram," which methods are to be utilized to determine that all control rods have been fully inserted into the core?

a. REFUEL MODE SELECT PERMISSIVE light ON **ONLY**.

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

c. All green FULL-IN lights on the full core display are ON **AND** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

d. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

ANSWER: 70 13673

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

Answer source: 2.1.5 p. 5, Attachment 1, step 1.5

Distractors:

a. Utilizing the green full in lights on the full core display is also allowed.
c. The use of the RPIS PMIS display is not directed by procedure 2.1.5.
d. The use of the RPIS PMIS display is not directed by procedure 2.1.5

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	16796	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320126, CNS Abnormal Procedures (RO) Cooling Water

Related Lessons	
INT0320126	CNS Abnormal Procedures (RO) Cooling Water

Related Objectives	
INT0320126Q0Q0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
295021.AA2.07	Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.10 / 43.5 / 45.13) Reactor recirculation flow (2.9/3.1)

QUESTION: 71 16796 (1 point(s))

With the plant shutdown in Mode 4 and RHR loop "B" operating in Shutdown Cooling, the following conditions exist:

- ➢ Reactor pressure is 0 psig
- ➢ Recirc suction temperature is 170°F
- ➢ Reactor water level is 36" (NR)
- ➢ RHR pumps "A" and "C" have both motors disconnected from their pumps

Subsequently, "B" RHR Loop develops a leak, requiring the Control Room operators to remove RHR loop "B" from service.

What action is required, and for what reason?

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

b. RPV water level must be raised to > 48" in order to maximize Reactor coolant contact with RPV metal for enhanced heat transfer to the Drywell atmosphere.

c. A Reactor Recirculation pump must be started to reduce the possibility of excessive thermal stresses on the CRD stub tubes.

d. A Reactor Recirculation pump must be started in order to reduce the possibility of thermal binding of the RHR-MO-25A/B valves caused by the expected coolant heatup.

ANSWER: 71 16796

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

Answer source: 2.4SDC p. 10, Attachment 2, step 1.1

Distractors:

b. Water level is raised to enhance natural ciruclation flow, not heat transfer.
c. A Recirculation pump is started, but not to prevent thermal stresses on the stub tubes. Starting a recirc pump causes thermal stress on the stub tubes.
d. A Recirculation pump is started, but not to prevent thermal binding of the gate valves. These valves are cycled if closed during a cooldown.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	14426	01	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum

Related Lessons	
INT0320132	CNS Abnormal Procedures (RO) Off Gas/Vaccuum
COR0011402	OPS MAIN TURBINE

Related Objectives	
COR0011402001060D	Given a specific Main Turbine and Auxilary systems malfunction, determine the effect on any of the following: Condenser vacuum
INT0320132L0L0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
COR0011402001140E	Given a Main Turbine and Auxiliaries component manipulation, predict and explain the changes in the following parameters: Steam seal pressure

Related References	
2.4VAC	Loss of Condenser Vacuum
2.2.75	Procedure 2.2.75, Steam Sealing System

Related Skills (K/A)	
295002.AK2.11	Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) Seal steam: Plant-Specific. (2.6/2.7)

QUESTION: 72 14426 (1 point(s))

The plant is at 30% power with the following conditions:

- Main Condenser vacuum is currently 27"Hg and degrading
- SJAE air flow (AR-FR-47) has shown a large increase over the last several minutes
- Gland Steam supply pressure indicator (MS-PI-83) indicates 0 psig

What action will correct the problem?

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

b. Throttle ***Closed*** MS-MO-BMV3, Steam Supply Bypass Valve.

c. Throttle ***Open*** MS-MO-BMV4, Steam Unloader Bypass Valve.

d. Throttle ***Closed*** MS-MO-BMV4, Steam Unloader Bypass Valve.

ANSWER: 72 14426

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

Regardless of the valve alignment or supply, the correct action to take at this power level is to open the steam bypass from main steam to put pressure on the seals.

Answer source: 2.2.75 p. 3, step 4.19.2

Distractors:

b. Taking action to close BMV3 could only make a low pressure situation worse, and BMV3 is probably already closed to begin with.
c. BMV4 would be opened if pressure was too high and steam was leaking from the seals.
d. Throttling closed BMV4 could make a high pressure situation worse, if the steam unloaders did not have enough capacity to maintain pressure.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	4214	00	05/21/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320134, OPS CNS Abnormal Procedures (RO) - 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).
INT0320134D0D0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.4.FIRE-SD	FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM

Related Skills (K/A)	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. (CFR: 43.5 / 45.13) (3.8/3.6)

QUESTION: 73 4214 (1 point(s))

A shutdown from outside the Control Room is in progress per 5.4FIRE-S/D.

How is Service Water flow established/controlled through RHR HX after the "B" and "D" Service Water Pumps have been started?

a. The Reactor Building operator manually operates SW-MO-89B from the handwheel.

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

c. The ASD Operator controls flow by the control switch for SW-MO-89B on the RHR panel in the ASD Room.

d. The Reactor Building operator fully opens SW-MO-89B at MCC-Y and the Control Building operator throttles on the SWBP discharge valve.

ANSWER: 73 4214

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

5.4FIRE-SD directs the operator to ensure breaker is closed and to rotate silver screw under switch which defeats mechanical interlock and open cubicle door. Then to remove control power fuse to prevent spurious operation, and to open valve by pressing button on LOWER contactor for time indicated for valve on the procedure attachment.

Answer source: 5.4FIRE-S/D p.22, step 1.1.6

Distractors:

a. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
c. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
d. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	5127	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	SKL0080101, WATCHSTANDING PRINCIPLES

Related Lessons
SKL0080101 WATCHSTANDING PRINCIPLES

Related Objectives	
SKL00801010011500	State what should be done if an out of limits reading is recorded on the logs or if an error is made recording a reading.

Related References	
1.9	Control and Retention of Records

Related Skills (K/A)	
2.1.18	Ability to make accurate / clear and concise logs / records / status boards / and reports. (CFR: 45.12 / 45.13) (2.9/3.0)

QUESTION: 74 5127 (1 point(s))

The RO realizes that an incorrect river level was transcribed by the Station Operator into 6.LOG.601, DAILY SURVEILLANCE LOG - MODES 1, 2, AND 3."

How is the the wrong river level corrected?

a. Correct number should be entered **AND** dated.

b. Circle the number, **AND** write in the correct number with an explanation.

c. Circle, initial **AND** date the old number, **AND** write in the correct number.

d. Draw one line through the incorrect old number, initial **AND** date, **AND** write in the correct number.

ANSWER: 74 5127

d. Draw one line through the incorrect old number initial **AND** date, **AND** write in the correct number.

Procedure 1.9 requires a single line to be drawn through the information to be corrected such that the information crossed out must remain legible. The person making the correction must initial and date the correction. Additional information may be added.

Distractors:

a. Required to draw a single line through correction, intial & date and write in the correct entry.
b. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.
c. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	2167	01	02/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0010502, FIRE PROTECTION SYSTEM

Related Lessons
COR0010502 FIRE PROTECTION SYSTEM

Related Objectives	
COR0010502001130F	Briefly describe the following concepts as they apply to the Fire Protection system: Principle of operation of the smoke detectors
COR0010502001110F	Given plant conditions, determine if the following should occur: Initiation of Total Flooding High Pressure CO2 in associated Diesel Generator Room.

Related References	
2.2.2	Procedure 2.2.2, Carbon Dioxide Systems

Related Skills (K/A)	
286000.K5.07	Knowledge of the operational implications of the following concepts as they apply to FIRE PROTECTION SYSTEM: (CFR: 41.5 / 45.3) Smoke detection (2.6/2.7)

QUESTION: 75 2167 (1 point(s))

What conditions must be satisfied to energize the DG room #1 high pressure CO_2 system cylinder solenoid valve with DG room #1 and #2 MAIN-RESERVE switches in MAIN?

a. One of four smoke detectors trip **AND** one of one thermal detector trips (after a time delay).

b. One of four smoke detectors trip **AND** one of one thermal detector trips (no time delay).

c. Two of four smoke detectors trip **OR** one of one thermal detector trips (no time delay).

d. Two of four smoke detectors trip **OR** one of one thermal detector trips (after a time delay).

ANSWER: 75 2167

d. Two of four smoke detectors trip **OR** one of one thermal detector trips (after a time delay).

The system is automatically actuated by any two smoke detectors in the DG room, and/or by a thermal detector, set at 190°F, in the associated diesel fuel oil day tank room.

Answer source: 2.2.2 p. 16, step 1.2.3

Distractors:

a. System actuation would result from the thermal detector ONLY without any smoke detectors tripped. Any automaitic actuation is preceded by a time delay.
b. System actuation would result from the thermal detector ONLY without any smoke detectors tripped. Any automaitic actuation is preceded by a time delay.
c. Although the logic indicated would actuate the system, automatic actuations are preceded by a 50 sec time delay.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	1290	01	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYSTEM

Related Lessons
COR0010602 FUEL POOL COOLING AND DEMINERALIZING SYSTEM

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

Related References	
2.2.32	Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System
2.4FPC	Fuel Pool Cooling Trouble

Related Skills (K/A)	
233000.A1.07	Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: (CFR: 41.5 / 45.5) System temperature (2.7/2.8)

QUESTION: 76 1290 (1 point(s))

What is HIGHEST fuel pool temperature expected under **maximum normal** heat load?

a. 110 °F.

b. 125 °F.

c. 150 °F.

d. 160 °F.

ANSWER: 76 1290

b. 125 °F.

Answer source: 2.2.32 p. 45, step 1.2.15

Distractors:

a. This is the upper end of the administrative limit for fuel pool temperature during normal operation.
c. This is the upper end of the expected fuel pool temperature with the core offloaded and the FPC system assisted by RHR.
d. This temperature is above the expected maximum for all FPC operations.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	5092	01	03/24/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	3	1	4	Multiple Choice	

Topic Area	Description
Systems	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYSTEM

Related Lessons
COR0010602 FUEL POOL COOLING AND DEMINERALIZING SYSTEM

Related Objectives	
COR0010602001070B	Given a specific FPC malfunction, determine the effect on any of the following: Pool/Rx Well Water Level

Related References	
2.4FPC	Fuel Pool Cooling Trouble
2.2.32	Procedure 2.2.32, Fuel Pool Cooling and Demineralizer System

Related Skills (K/A)	
295023.AA1.02	Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) Fuel pool cooling and cleanup system (2.9/3.1)

QUESTION: 77 5092 (1 point(s))

Given the following conditions:

- The Fuel Pool Cooling (FPC) System is operating with one pump **AND** one heat exchanger in service
- The Fuel Pool Gates are installed
- A leak occurs on the outlet pipe of the Skimmer Surge Tanks
- **NO** operator action is taken

What effect will these conditions have on FPC System cooling capability **AND** Fuel Pool water level?

FPC System cooling capability . . .

a. **AND** Fuel Pool water level will be unchanged.

b. will be lost **AND** Fuel Pool water level will continuously lower.

c. will be lost **AND** Fuel Pool water level will lower slightly **AND** stabilize.

d. will be unchanged **AND** Fuel Pool water level will lower slightly **AND** stabilize.

ANSWER: 77 5092

c. will be lost **AND** Fuel Pool water level will lower slightly **AND** stabilize.

Water from the pool overflows the weirs into the Skimmer Surge Tanks and is then pumped from the Skimmer Surge Tanks back to the pool. Since makeup to the system is manual, a leak will drain the tanks. Low level in the tanks will trip the pumps . With no water being pumped to the pool, level will lower until it reaches the top of the weirs and then stabilize.

<u>Answer source:</u> 2.4FPC p. 1, step 2.1

Distractors:

a. The pump trips and pool level will stabilize at the top of the weirs.
b. Pool level will stabilize at the top of the weirs.
d. The pump trips.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	3724	02	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0010802, 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	Procedure 2.2.84, HVAC Main Control Room and Cable Spreading Room

Related Skills (K/A)	
600000.AA1.05	Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Plant and control room ventilation systems (3.0/3.1)

QUESTION: 78 3724 (1 point(s))

A fire has occurs in the Cable Spreading Room with the plant operating at 100% power. The fired does not spread beyond the Cable Spreading Room.

What impact does the fire have on the Control Room Ventilation System?

a. 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.

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 trip. The Emergency Bypass Train does **NOT** start.

d. 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.

ANSWER: 78 3724

c. Fire Dampers isolate the Control Room. The Supply Fans trip. The Emergency Bypass Train does **NOT** start.

Answer source: COR001-08-01 p. 63
COR001-08-01 figure 23
2.2.84 p. 25, Attachment 4

Distractors:

a. The control room is isolated. The supply fans trip. The emergency bypass train does not start.
b. The control room is isolated..
d. Control room, not Cable Spreading room. The supply fans do trip. The Bypass Train does not start.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	14041	00	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0011602, Off Gas

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001090D	Explain the following Off Gas system related concepts: Charcoal absorption of fission product gases
COR0011602001010O	State the purpose of the following items related to Off Gas system: Charcoal Absorber Beds

Related References	
2.2.58	Procedure 2.2.58, AOG System

Related Skills (K/A)	
2.1.28	Knowledge of the purpose and function of major system components and controls. (3.2/3.3)

QUESTION: 79 14041 (1 point(s))

Why are the Off-gas system charcoal beds maintained at sub-freezing temperatures?

The low temperatures . . .

a. freeze any remaining moisture in the Off-gas stream to prevent it's intrusion onto the charcoal.

b. increase the adsorption coefficients of Krypton and Xenon increasing selective adsorption and retention.

c. reduces the relative humidity of the Off-gas stream in the charcoal beds increasing adsorption of Krypton and Xenon.

d. increases density of the gases in the Off-gas stream, thereby reducing the volume and increasing the stream contact time with the charcoal bed.

ANSWER: 79 14041

b. increase the adsorption coefficients of Krypton and Xenon increasing selective adsorption and retention.

A lower than ambient operating temperature of 0°F is selected as the adsorption coefficients (K) of krypton and xenon increase with a decrease in temperature.

Answer source: COR001-16-01, p. 36, section 4

Distractors:

a. Freezing water in or on the charcoal bed would not improve its ability to adsorb Xe and Kr. In addition the relative humidity of the off gas stream should be sufficiently below the temperature of the charcoal beds.
c. The relative humidity of the off-gas stream is controlled by components upstream of the charcoal beds.
d. Density changes (and therefore volume) alone would not significantly affect the flow rate through the charcaol beds.

2.1.28 Knowledge of the purpose and function of major system components and controls. as it applies to 271000 Offgas System

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	3973	01	02/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0011602, Off Gas

Related Lessons
COR0011602 Off Gas

Related Objectives	
COR0011602001080G	Describe the Off Gas system design feature(s) and/or interlock(s) that provide for the following: Automatic system isolation

Related References	
2.3_B-3	PANEL B - ANNUNCIATOR B-3

Related Skills (K/A)	
256000.K4.10	Knowledge of REACTOR CONDENSATE SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Non-condensable gas removal (2.7/2.7)

QUESTION: 80 3973 (1 point(s))

The plant was operating at power when an explosion occurred in the Off-Gas system. The SJAE suction valves automatically closed. The Off-Gas system sustained no damage during the transient and the ERP monitors spiked but immediately returned to normal. Hold up pipe temperature **AND** pressure also returned to normal values.

Which of the following describes the process that reopens the SJAE suction valves?

The Off-Gas High Pressure AND Temperature signals ____________ , and the SJAE suction valves __________________.

a.	reset automatically	are manually reopened
b.	reset automatically	automatically reopen
c.	must be reset manually	are manually reopened
d.	must be reset manually	automatically reopen.

ANSWER: 80 3973

c. must be reset manually are manually reopened

The isolation valves are re-opened manually after the isolation signal is clear and the logic is reset using either the SJAE Suction Isol Reset pushbutton on Control Room Panel B or Local Panel 1R-1E.

Answer source: COR001-16-01, p. 20, section 6.b
COR001-16-01, p. 56, Main Control Room Controls Table

Distractors:

a. The isolation signals do not reset automatically.
b. The isolation signals do not reset automatically and the SJAE suction valves do not open automatically.
d. The SJAE suction valves do not reopen automatically.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	3085	02	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020102, AVERAGE POWER RANGE MONITOR

Related Lessons	
COR0020102	AVERAGE POWER RANGE MONITOR
SKL0124201	AVERAGE POWER RANGE MONITOR SYSTEM

Related Objectives	
COR0020102001080C	Describe the APRM design feature(s) and/or interlock(s) that provide for the following: Alarm seal-in
COR0020102001110E	Given an Average Power Range Monitor System control manipulation, predict the changes in the following parameters: Lights and alarms
SKL012420100A030F	Given plant conditions, predict changes in the following APRM system component/parameters: Lights/alarms.
SKL012420100B030F	Manually operate the APRM system to control the following: Lights/alarms.

Related References	
6.1APRM.303	APRM System Channel Functional Test (Mode Switch In Run/Div 1)

Related Skills (K/A)	
215005.A2.01	Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or...: (CFR: 41.5/45.6) Power supply degraded (2.7/3.1)

QUESTION: 81 3085 (1 point(s))

The plant was operating at rated power when the power supply fuses for "B" APRM blew. The fuses have been replaced and the 1/2 scram has been reset.

Given that annunciator 9-5-1/B-8 **APRM UPSCALE** clears, what actions would the operator need to perform (if any) to clear the alarms on panel 9-14 **AND** clear the alarms on panel 9-5?

The alarms lights on panel 9-14 ________________ and the alarm lights on panel 9-5 benchboard ________________.

a.	clear automatically	clear automatically
b.	clear automatically	must be reset from 9-5
c.	must be reset from 9-14	clear automatically
d.	must be reset from 9-14	must be reset from 9-5

ANSWER: 81 3085

c.	must be reset from 9-14	clear automatically

Answer source: 6.1APRM.303, p. 5, step 4.19
COR002-01-02, p. 20, Control Room Controls table
COR002-01-02, p. 14, section a)
COR002-01-02, p. 14, section 2.

Distractors:

a. The alarms lights on panel 9-14 must be reset from 9-14.
b. The alarms lights on panel 9-14 must be reset from 9-14. The alarms lights on panel 9-5 clear automatically.
d. The alarms lights on panel 9-5 clear automatically.

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	18311	01	02/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020202, OPS CONDENSATE AND FEED

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001120C	Given plant conditions, determine if: Minimum Flow Valves should have repositioned

Related References	
2.2.28.1	Procedure 2.2.28.1, Feedwater System Operation
2.2.28	Procedure 2.2.28, Feedwater System Startup And Shutdown

Related Skills (K/A)	
259001.K4.03	Knowledge of REACTOR FEEDWATER SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) RFP minimum flow (2.7/2.7)

QUESTION: 82 18311 (1 point(s))

A plant startup is in progress, and RFP 1A is the second RFP to be started. As RFP is being placed in service, RFP 1A flow quickly rises from less than 1000 gpm to approximately 3000 gpm, but then, after approximately 1 minute, RFP 1A experiences excessive vibration and flow is lowered, stabilizing at approximately 100 gpm.

What describes the expected RFP 1A minimum flow valve response?

The RFP minimum flow valve _____________ as flow rises; then, the minimum flow valve _____________ as flow lowers.

a. automatically closes automatically reopens

b. automatically closes must be manually reopened

c. must be manually closed automatically reopens

d. must be manually closed must be manually reopened

ANSWER: 82 18311

c. must be manually closed automatically reopens

Each Reactor Feed pump (RFP) is equipped with an air operated minimum flow valve which will automatically open if pump flow decreases to approximately 2000 gpm. These valves have no automatic closing feature and must be closed, using the minimum flow c/s on Panel A, when pump discharge flow is greater than 2000 gpm.

Answer source: COR002-02-02, p. 30, section 5

Distractors:

a. The minimum flow valve doesn't automatically close as flow increases.
b. The minimum flow valve does not close automatically as flow increases and the valve automatically reopens if flow decreases.
d. The minimum flow valve automatically reopens on decreasing flow.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	14053	00	03/31/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	RO only COR0020202, OPS CONDENSATE AND FEED

Related Lessons
COR0020202 OPS CONDENSATE AND FEED

Related Objectives	
COR0020202001080A	Predict the consequences a malfunction of the following would have on the Condensate and Feedwater system: Plant Air

Related References	
5.2AIR	Loss of Instrument Air

Related Skills (K/A)	
300000.K3.02	Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: (CFR: 41.7 / 45.6) Systems having pneumatic valves and controls (3.3/3.4)

QUESTION: 83 14053 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 100% power
- ➢ The air supply line to COND BOOSTER P B MIN FLOW, MC-AOV-FCV10 is severed
- ➢ **NO** operator action is taken

How will the Feed System respond to these conditions over the next one (1) minute?

"B" Condensate Booster Pump Minimum Flow Valve (MC-AOV-FCV10) fails . . .

a. closed. The speed of **BOTH** RFPs rise.

b. closed. The speed of **BOTH** RFPs remains unchanged.

c. open. The speed of **BOTH** RFPs rise.

d. open. The speed of **BOTH** remains unchanged.

ANSWER: 83 14053

c. open. The speed of **BOTH** RFPs rise.

MC-AOV-FCV10 fails open on loss of air (see attachment 1 of 5.2AIR). When the valve opens, feed flow will lower causing a FF-SF mismatch. RFP speed will rise until FF=SF and level is at 35" or the Reactor scrams on low level.

Answer source: COR002-02-02, p. 81, section 4)

Distractors:

a. MC-AOV-FCV10 fails open on loss of air.
b. MC-AOV-FCV10 fails open on loss of air.
d. When the valve opens, feed flow will lower causing a FF-SF mismatch. RFP speed will rise until FF=SF and level is at 35".

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	3302	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020302, CONTAINMENT

Related Lessons
COR0020302 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	
2.2.40	Procedure 2.2.40, HVAC Drywell Cooling

Related Skills (K/A)	
223001.A3.03	Ability to monitor automatic operations of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES including: (CFR: 41.7 / 45.7) System indicating light and alarms (3.4/3.3)

QUESTION: 84 3302 (1 point(s))

A small break LOCA has occurred with the following conditions:

➢ Reactor water level	+45" (stable NR)
➢ Reactor pressure	560 psig (lowering at 10 psig per minute)
➢ Drywell pressure	3.1 psig (rising at 0.1 psig per minute)
➢ Drywell temperature	195°F (rising at 2°F per minute)
➢ "C" Drywell FCU control switch	RUN

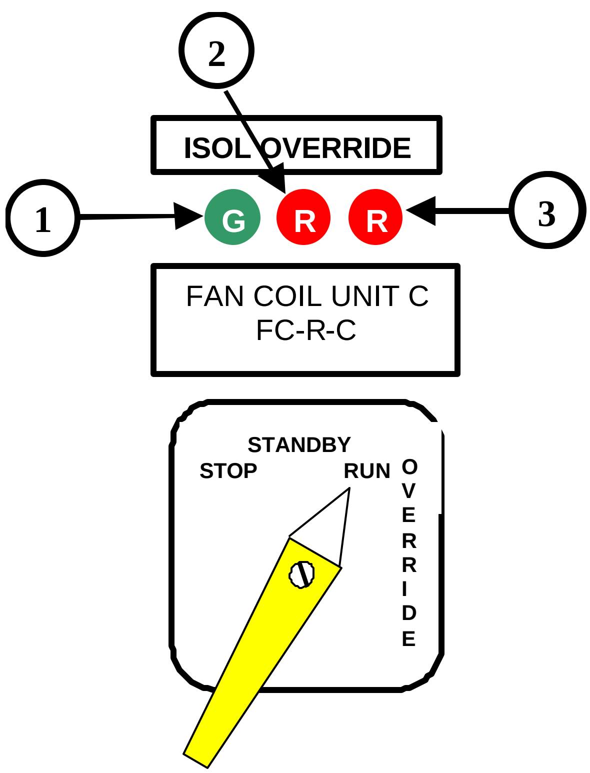
Which lamp(s) is/are illuminated (1, 2 or 3) for "C" Drywell Fan Coil Unit (FCU) and what actions are required for "C" Drywell FCU to operate 10 minutes from now?

a. Lamp #1 is illuminated. Place FCU control switch in OVERRIDE.

b. Lamp #1 is illuminated. Install jumpers to bypass the Core Spray injection valve low pressure permissive signal.

c. Lamp #2 is illuminated. Place FCU control switch in OVERRIDE.

d. Lamp #2 and #3 are illuminated. Install jumpers to bypass the high drywell pressure signal and place FCU control switch in OVERRIDE.



ANSWER: 84 3302

a. Lamp #1 is illuminated. Place FCU control switch in OVERRIDE.

Answer source: COR002-03-02, p. 26

Distractors:

b. The core spray low pressure permissive signal does not affect FCU logic, just the injection valves.
c. The FCU is not running.
d. The FCU is not in override and is not running. No need to bypass the high drywell pressure signal.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	14038	00	03/09/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020402, CONTROL ROD DRIVE HYDRAULICS

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

Related Objectives	
COR0020402001120C	Given a specific CRDH system malfunction, determine the effect on any of the following: Control rod drive mechanisms (CRDMs)
COR0020402001120D	Given a specific CRDH system malfunction, determine the effect on any of the following: Reactor water cleanup pumps
COR0020502001090A	Predict the consequences a malfunction of the following would have on the CRDMs: Loss of CRDH Pumps
COR0020402001120A	Given a specific CRDH system malfunction, determine the effect on any of the following: Recirculation pumps

Related References	
2.2.8	Procedure 2.2.8, Control Rod Drive System
2.4CRD	CRD TROUBLE

Related Skills (K/A)	
201003.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM: (CFR: 41.7 / 45.7) Control rod drive hydraulic system (3.3/3.3)

QUESTION: 85 14038 (1 point(s))

A plant startup is in progress with reactor pressure at 926 psig and reactor power at 8%. The "A" CRD pump trips and cannot be restarted. All attempts by the crew to start "B" CRD pump are also unsuccessful.

If operation were to continue with these conditions, which of the following would result?

a. reduced CRDM seal life

b. inability to scram control rods

c. reactor recirculation (RR) pump seal failure

d. exceed the acceptable pressure temperature range for RWCU pump operation

ANSWER: 85 14038

a. reduced CRDM seal life.

The loss of both CRD pumps results in a loss of cooling to the CRDMs Operation for long periods at high temperatures will reduce the life of the CRDM seals.

Answer source: COR002-04-02, p.31, section b

Distractors:

b. With reactor pressure above 900 psi normal scram times will be met even if accumulator becomes reduced by the loss of CRDH pumps.
c. Long term the seal life of the RR pump may be shortened but the seal should remain sufficiently cooled by REC to the jacket around the seal.
d. Loss of CRDH would not impact reactor pressure or RWCU inlet temperature.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	14496	00	05/09/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020702, OPS DC ELECTRICAL DISTRIBUTION

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

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

Related References	
3058	DC One Line Diagram
2.2.24.1	250 VDC ELECTRICAL SYSTEM (DIV 1)

Related Skills (K/A)	
295004.AA1.01	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.7 / 45.6) D.C. electrical distribution systems (3.3/3.4)

QUESTION: 86 14496 (1 point(s))

The plant is at 100% power with RCIC in full-flow test mode when annunciator C-1/C-1, 250 VDC BATT CHARGER 1A TROUBLE alarms.

- CRT alarm message indicates:

  (3708) 250 VDC BATTERY CHARGER 1A DC VOLTAGE HIGH (in and reset)
  (3707) 250 VDC BATTERY CHARGER 1A AC VOLTAGE FAILURE (in and reset)
  (3709) 250 VDC BATTERY CHARGER 1A DC VOLTAGE LOW

The following 250 VDC indications are observed:

- 250 VDC Bus 1A indicates 250 volts and stable
- 250 VDC Battery 1A indicates 75 amps out and stable
- 250 VDC Charger 1A indicates 0 amps

What is the status of 250 VDC electrical distribution?

a. The AC input breaker on the 250V charger has tripped open automatically.

b. The DC output breaker on the 250V charger has tripped open automatically.

c. The 150 amp fuse on the feeder to the 250 VDC RCIC Starter Rack has blown.

d. The 300 amp fuse on the feeder from 250 VDC BATT CHARGER 1A has blown.

ANSWER: 86 14496

a. The AC input breaker on the 250V charger has tripped open automatically. DC output over voltage causes the AC input breaker on a 250V CHARGER to trip.

Answer source: COR002-07-02, p. 15, section 2

Distractors:

b. The DC output breaker does NOT automatically trip open.
c. If the 150 amp fuse on the feeder to the 250 VDC RCIC Starter Rack had blown, Annunciator C-1/A-1, 250 VDC SWGR BUS 1A BLOWN FUSE would have alarmed and CRT alarm message would indicate (3703) RCIC Starter Rack normal feeder; also, battery 1A would not indicate being loaded (75 amps and stable) as the RCIC gland seal vacuum pump and condensate pump are currently the only energized loads on 250 VDC Bus 1A.
d. If the 300 amp fuse on the feeder from 250 VDC BATT Charger 1A had blown, Annunciator C-1/A-1, 250 VDC SWGR BUS 1A BLOWN FUSE would have alarmed and CRT alarm message would indicate (3702) Feeder from Battery Charger A.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	14034	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0020802, DIESEL GENERATORS

Related Lessons
COR0020802 DIESEL GENERATORS

Related Objectives	
COR0020802001110F	Predict the consequences a malfunction of the following would have on the Diesel Generators: DC Power

Related References	
COR0020802	Diesel Generators

Related Skills (K/A)	
264000.K1.02	Knowledge of the physical connections and/or cause- effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) D.C. electrical distribution (3.3/3.4)

QUESTION: 87 14034 (1 point(s))

The plant was operating at 100% power when a loss of off-site power occurred. Both diesel generators automatically started and loaded their respective busses. An electrical fault results in the loss of 125 VDC panel DG1.

How is DG1 affected and why?

DG1 . . .

a. stops due to loss of power to the fuel oil booster pump.

b. stops due to loss of power to the the electronic trip solenoid.

c. continues to run but the day tank cannot be refilled due to loss of power to transfer pumps.

d. continues to run but the fuel oil booster pump is unavailable should the engine driven pump fail to develop sufficient fuel oil pressure.

ANSWER: 87 14034

b. stops due to loss of power to the the electronic trip solenoid.

Since both the electronic trip valve (20SD) energizes on the auto start to position the fuel control cylinder to the "FUEL ON" position, a loss of 125 VDC to this valve causes the Diesel Generator to trip.

Answer source: COR002-08-02, p. 26

Distractors:

a. Although a loss of power to the fuel oil booster pump has occurred this is not the reason the DG stops. The engine driven pump would be sufficient to supply the fuel oil needs of the DG.
c. The DG does not continue to run.
d. Although the fuel oil booster pump is in fact unavailable to backup the engine driven fuel oil pump, the DG does not continue to run due to the loss of power to the electronic trip valve.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	3286	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0021502, NUCLEAR BOILER INSTRUMENTATION

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001020F	Describe the interrelationships between NBI and the following: Main Turbine/Feedwater
COR0021502001060E	Given a specific NBI malfunction, determine effect on any of the following: Main turbine and feedwater
COR0021502001040A	Briefly describe the following concepts as they apply to NBI: Vessel level measurement
COR0021502001050A	Predict the consequences of the following on the NBI: Detector equalizing valve leaks

Related References	
4.6.1	Procedure 4.6.1, Reactor Vessel Water Level Indication

Related Skills (K/A)	
216000.K1.16	Knowledge of the physical connections and/or cause- effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Main turbine (3.0/3.1)

QUESTION: 88 3286 (1 point(s))

The plant is operating at 100% power with NBI-LT-52C level transmitter (Narrow Range Reactor Water level instrument) failed upscale. NBI-LT-52B level transmitter is selected for level control.

Prior to isolating the NBI-LT-52C level transmitter from service for maintenance, the equalizing valve for NBI-LT-52C is inadvertently opened by I&C.

Assume **NO** operator actions are taken.

Which one of the following describes **ONLY** the effects **DIRECTLY** produced by these conditions?

a. A high reactor water level alarm is received.

b. A full scram is received on a low RPV water level signal.

c. The RFPs and the Main Turbine will trip on a high RPV water level signal.

d. A ½ scram is received on RPS trip system "A" due to a low RPV water level signal.

ANSWER: 88 3286

c. The RFPs and the Main Turbine will trip on a high RPV water level signal.

Opening LT-52C equalization valve will drain the reference leg and will result in "A" channel indicating upscale as well, giving a high level trip to the RFPT and the MT with the subsequent full scram.

Answer source: COR002-15-02, p. 44, section 6
COR002-15-02, p. 35, Interlocks and Trips

Distractors:

b. A high water level alarm will not occur as "B" is selected.
c. The scram will be due to the turbine trip.
d. A full scram is received on low level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	1238	00	03/24/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001150E	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod drift alarm

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
214000.A2.02	Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those...: (CFR: 41.5 / 45 Reactor SCRAM (3.6/3.7)

QUESTION: 89 1238 (1 point(s))

The plant was operating at rated power when a manual reactor scram was inserted. The following conditions exist:

- ➢ All control rods have fully inserted
- ➢ The reactor mode switch has been placed in SHUTDOWN
- ➢ The scram has not been reset

What are the **MINIMUM** actions necessary to reset the control rod drift indications on the full core display (vertical section of panel 9-5)?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

b. Reset the reactor scram. After control rods have settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

c. Reset the reactor scram. Select each control rod with a drift alarm. After control rod has settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET. Repeat for each control rod with a drift alarm.

d. Select each control rod with a drift alarm. Momentarily place ROD MOVEMENT CONTROL switch to OUT NOTCH. After control rod has settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET. Repeat for each control rod with a drift alarm.

ANSWER: 89 1238

b. Reset the reactor scram. After control rods have settled at position 00, momentarily rotate ROD DRIFT ALARM TEST switch to RESET.

Answer source: 2.1.5 p. 4, step 5.10

Distractors:

a. The scram must be reset.
c. The control rods do not need to be selected.
d. The scram must be reset and do not need to notch each rod.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	2816	01	03/21/2003	05/21/2003	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	RO only COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030N	Describe RHR System design feature(s) and/or interlocks which provide for the following: Prevention of leakage to the environment through system heat exchanger
COR0022302001040G	Describe the interrelationship between the RHR system and the following: RHR Service Water
COR0022302001080R	Predict the consequences a malfunction of the following will have on the RHR system: RHR Service Water

Related References	
2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1

Related Skills (K/A)	
205000.K6.08	Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): (CFR: 41.7 / 45.7) RHR service water: Plant-Specific (3.5/3.7)

QUESTION: 90 2816 (1 point(s))

The plant is in the process of cooling down. The plant shutdown was required due to fuel failure. The following conditions exist:

- ➢ "A" RHR pump is in Shutdown Cooling
- ➢ RPV water level is + 50" (NR)
- ➢ RPV pressure is maintained at 20 psig using the Main Turbine Bypass Valves
- ➢ It is determined that tubes in the "A" RHR Heat Exchanger have completely failed

What is the consequence of continuing to operate "A" RHR pump in Shutdown Cooling?

a. RPV pressure will rise.

b. RPV water level will rise.

c. Radioactive water will be released to the environment.

d. Radioactive materials will be deposited in Reactor Building sumps.

ANSWER: 90 2816

b. RPV water level will rise.

Answer source: COR002-23-02, p. 22, section 4

Distractors:

a. Service water flow into the RPV will lower RPV pressure. The RPV will never go solid with the MSIVs open (BPVs in use for pressure control requires MSIVs be open).
c. Service water pressure is higher than RHR pressure.
d. Service water pressure is higher than RHR pressure, there is no release path to the Reactor Building.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	19100	1	03/20/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only SKL0124223, OPS RESIDUAL HEAT REMOVAL

Related Lessons	
COR0022302	RESIDUAL HEAT REMOVAL
SKL0124223	OPS RESIDUAL HEAT REMOVAL

Related Objectives	
SKL012422300A0200	Explain the Residual Heat Removal system limitations and precautions as stated in the SOP 2.2.69, SOP 2.2.69.1, SOP 2.2.69.2 and SOP 2.2.69.3.
COR0022302001050B	Briefly describe the following concepts as they apply to the RHR system: Valve operation

Related References	
2.2.69.3	Procedure 2.2.69.3, RHR Suppression Pool Cooling And Containment Spray
2.0.1	Procedure 2.0.1, Plant Operations Policy

Related Skills (K/A)	
203000.A4.02	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) System valves (4.1*/4.1*)

QUESTION: 91 19100 (1 point(s))

When operating RHR-MO-66B, B HX BYPASS VLV, what precaution applies and what is the bases for this precaution?

a. Do **NOT** hold the control switch in <u>OPEN</u> after the **red** indicating light turns **ON** to prevent tripping the valve breaker.

b. Do **NOT** hold the control switch in <u>CLOSE</u> after the **green** indicating light turns **ON** to prevent hammering the valve.

c. Hold the control switch in <u>CLOSE</u> for 5 seconds after the **red** indicating light turns **OFF** to ensure the valve closure is terminated by torque switch.

d. Hold the control switch in <u>OPEN</u> for 3 seconds after the **green** indicating light turns **OFF** to ensure the valve opening is terminated by torque switch.

ANSWER: 91 19100

c. Hold the control switch in <u>CLOSE</u> for 5 seconds after the **red** indicating light turns **OFF** to ensure the valve closure is terminated by torque switch.

RHR-MO-66B, B HX BYPASS VLV is a throttle valve and throttle valves are held in closed an additional 5 seconds to ensure they are closed by their torque switch.

<u>Answer source:</u> 2.0.1 p. 6, step 9.1

Distractors:

a. This is based on the precaution from the procedure on seal-in limitorque operators.
b. This is based on the precaution from the procedure on seal-in limitorque operators.
d. The valve is held for 5 seconds (3 seconds is time delay for reversing directions) to prevent terminating the closure with the close limit switch.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	5401	01	03/17/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0022402, ROD BLOCK MONITOR

Related Lessons
COR0022402 ROD BLOCK MONITOR

Related Objectives
COR0022402001100A State the electrical power supplies to the following: RBM channels.

Related References	
4.1.5	Procedure 4.1.5, Rod Block Monitor System

Related Skills (K/A)	
215002.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM: (CFR: 41.7 / 45.7) RPS: BWR-3,4,5 (3.0/3.2)

QUESTION: 92 5401 (1 point(s))

Given the following conditions:

- ➢ The plant is operating at 88% power
- ➢ All APRM gain adjustment are within Technical Specification requirements
- ➢ RPS Panel PP1B power is lost
- ➢ Rod 22-03 is then selected

What effect does this have on control rod withdrawal blocks from the RBM system?
(Do not consider potential control rod withdrawal blocks from other systems.)

A control rod withdrawal block is . . .

a. present from **BOTH** RBM trip systems.

b. present from **only** RBM trip system "A".

c. present from **only** RBM trip system "B".

d. **NOT** present from **EITHER** RBM trip system.

ANSWER: 92 5401

d. **NOT** present from **EITHER** RBM trip system.

When RBM B deenergizes it generates a rod block. However, an edge rod is selected which bypasses all RBM rod blocks.

Answer source: COR002-24-02 p. 21, section b.(1)

Distractors:

a. Neither trip system will provide a rod block with an edge rod selected. "A" RBM would not cause a rod block even with the rod selected.
b. Neither trip system will provide a rod block with an edge rod selected.
c. Neither trip system will provide a rod block with an edge rod selected.

Source: Modified

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	5070	02	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	RO only COR0022902, STANDBY LIQUID CONTROL

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001100G	Predict the consequences a malfunction of the following would have on the SLC system: Tank Heaters

Related References	
2.2.74	Procedure 2.2.74, Standby Liquid Control System

Related Skills (K/A)	
211000.K5.07	Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: (CFR: 41.5 / 45.3) Tank heater operation (2.7/2.9)

QUESTION: 93 5070 (1 point(s))

Loss of the SLC Storage tank heaters can result in . . .

a. increased concentration of tank SLC solution.

b. SLC pump suction piping temperature decreases.

c. SLC solution temperature decreasing below saturation.

d. indicated tank level decreasing to less than actual level.

ANSWER: 93 5070

c. SLC solution temperature decreasing below saturation.

The loss of the SLC Storage Tank heater would result in a decreasing temperature of the SLC solution. This reduction in temperature could eventually result in SLC solution decresing to below saturation for the concentration in the tank.

Answer source: COR002-29-02 p. 19, section 3

Distractors:

a. The concentration of tank SLC solution should remain unchanged until the temperature goes below saturation, once it is below saturation temperature the sodium pentaborate could come out of solution resulting in a decreased concentration of the remaining liquid solution.
b. SLC pump suction piping temperature should remain relatively constant as the heat trace on the pump suction piping is unaffected.
d. Indicated tank level would not experience any appreciable effects. Even though the temperature of the solution would decrease, the mass (and therefore the weight of the solution) above the bubbler would remain constant. The decrease in temperature would decrease actual level as the density of the fluid increased.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	16441	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	RO only INT0070501, OPS Introduction to Technical Specifications

Related Lessons	
INT0070510	CNS Tech. Spec. 3.9, Refueling Operations
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT0070501001030A	From memory, define the following terms: Core Alteration
INT00705100010100	Given a set of plant conditions, recognize non-compliance with a Section 3.9 LCO.

Related References	
1.1	Definitions
3.9	Refueling Operations

Related Skills (K/A)	
2.2.27	Knowledge of the refueling process. (CFR: 43.7 / 45.13) (2.6/3.5)

QUESTION: 94 16441 (1 point(s))

Refueling operations are in progress with the reactor vessel head removed and a partial load of fuel is in the vessel. Shutdown margin check has been performed.

What action is a CORE ALTERATION?

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

b. Drive a Source Range Monitor detector to full in.

c. Perform a friction test on a control rod in a loaded cell.

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

ANSWER: 94 16441

c. Performing a friction test on a control rod in a loaded cell.

Core alteration includes movement of any reactivity controlling component with the exceptions specified: SRM movement is an exception, control rod movement if there is no fuel in the associated core cell is an exception. The LPRM instrument handling tool is not a reactivity control component.

Answer source: Technical Specifications definitions, p. 1.1-2

Distractors:

a. Control rod movement provided there are no fuel assemblies in the associated core cell is not considered to be a CORE ALTERATION.
b. Movement of a SRM is not considered to be a CORE ALTERATION.
d. The LPRM instrument handling tool is not a reactivity control component.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	14003	00	03/18/2003	06/13/2003	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
Technical Specifications, ODAM, TRM	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.

Related References	
3.3.1.1	Reactor protection system (RPS) instrumentation
3.3.1.1-1	Table - Functions 1a and 1b

Related Skills (K/A)	
295020.AA2.03	Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.10 / 43.5 / 45.13) Reactor power. (3.7/3.7)

QUESTION: 95 14003 (1 point(s))

The plant has been operating at rated power for 2 months following a refueling outage. An LPRM calibration has just been completed. I&C asked for the TIP detector to be run through channel 10 again. As the TIP being withdrawn (currently between the bottom of the RPV and the bottom of the core), the shear valve is inadvertently fired and the TIP cable breaks. All attempts to subsequently remove the broken part of the TIP cable have failed. The TIP ball valve has been sucessfully closed.

What is the effect on continued power operation and why?

Power operation may continue . . .

a. for the next 115 days (when including any allowed extensions) due to the need to declare APRMs inoperable.

b. until the next refueling outage. 3D MONICORE can compensate for LPRM detector aging without running TIP traces.

c. for only the next hour as all ARPMs must be immediately declared inoperable due to the inability to perform a required surveillance.

d. until core exposure has increased by 1250 MWD/T (when including any allowed extensions) due to the need to declare APRMs inoperable.

ANSWER: 95 14003

d. until core exposure has increased by 1250 MWD/T (when including any allowed extensions) due to the need to declare APRMs inoperable.

SR 3.3.1.1.8 will not be met at 1250 MWD/T, requiring all APRMs to be declared inoperable.

Answer source: SR 3.3.1.1.8, SR 3.0.2

Distractors:

a. Based on the other 92 day surveillance plus 25% extension (unaffected).
b. Shutdown is required at 1250 MWD/T. 3D MONICORE cannot substitue for LPRM detector surveillances.
c. The surveillance is "met" even if it cannot be "performed" for the next 1250 MWD/T.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.1.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	14026	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation

Related Lessons	
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705040010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.3 LCO, determine the ACTIONS that are required.
INT00705040010100	Given a set of plant conditions, recognize non-compliance with a Section 3.3 Requirement.

Related References	
3.3.3.1	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)

QUESTION: 96 14026 (1 point(s))

The plant is operating at rated power with the following conditions:

- ➢ NBI-PR-85A (Wide Range Reactor Pressure) (Post Accident Monitor) becomes inoperable at 1100 on 3/28
- ➢ NBI-LI-85A (Wide Range RPV water level) becomes inoperable at 0900 on 3/29
- ➢ NBI-PR-85B (Wide Range Reactor Pressure) (Post Accident Monitor) becomes inoperable at 1500 on 4/1

IF conditions do not change, what is the LATEST time that the plant is ***allowed*** 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: 96 14026

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.

Source: *New*

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	19123	00	05/31/2002	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	RO only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

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

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

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295035.EK1.01	Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10) Secondary containment integrity. (3.9/4.2*)

QUESTION: 97 19123 (1 point(s))

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

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

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

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

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

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

ANSWER: 97 19123

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

Answer source: INT008-06-17, p. 6, section C.1

Distractors:

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

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	16466	01	03/19/2003	06/13/2003	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
Procedures	RO only INT0320104, CNS Administrative Procedures General Operating Procedures

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

Related Objectives	
INT032010400A0100	Discuss Precautions and Limitations outlined in General Operating Procedure 2.1.1, Startup Procedure.

Related References	
2.1.1	Procedure 2.1.1, Startup Procedure

Related Skills (K/A)	
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: 45.1) (3.7/3.6)

QUESTION: 98 16466 (1 point(s))

During a reactor startup and heatup, reactor period was infinity after withdrawing a control rod. The following conditions are present with **NO** control rod movement for the last two (2) minutes:

- ➢ The reactor is on range 5 of the IRMs (rising)
- ➢ Reactor period is +120 seconds (shortening)
- ➢ Reactor coolant temperature is 180°F (rising)

Which one of the following represents the MINIMUM actions *required*?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. If reactor period shortens to 50 seconds, insert the last withdrawn control rod until period is longer than 50 seconds.

b. Range IRMs as necessary to keep them on scale until the Point of Adding Heat (POAH) is reached.

c. Insert control rods in reverse order to make the reactor subcritical.

d. Bypass the RWM and insert emergency power reduction rods (10.13 Att. 7) to position 00.

ANSWER: 98 16466

c. Insert control rods in reverse order to make the reactor subcritical.

Step 2.20 of procedure 2.1.1 states that conservative action ***is required*** whenever an unexpected situation arises with respect to reactivity, criticality, power level, or any other anomalous behavior of reactor core. This conservative action should include rod insertion to reduce power or a reactor scram without hesitation whenever such unanticipated or anomalous behavior is encountered. In this case indicated power is below the POAH yet temperature is rising and even with this negative reactivity feedback power is rising and period is getting shorter. All of which are significant indications of a significant anomoly.

Answer source: 2.1.1 p. 4, step 2.20

Distractors:

a. While the administrative limit for period is 50 seconds, the reactor is currently exhibiting anomolous behavior.
b. This action is not conservative, this action would allow the anomolous reactor behaviour to continue.

d. At this point in the startup operation would be below the 80% rod line and the emergency power reduction control rods are not available.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	13406	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	2	1	3	Essay	

Topic Area	Description
Abnormal/Emergency Procedures	RO only INT0320124, CNS Abnormal Procedure (RO) Reactor Recirculation

Related Lessons	
INT0320124	CNS Abnormal Procedure (RO) Reactor Recirculation

Related Objectives	
INT032012400F0F00	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).
INT032012400G0G00	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).
INT032012400H0H00	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	
2.4RR	REACTOR RECIRCULATION ABNORMALS

Related Skills (K/A)	
202001.A2.04	Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of...: (CFR: 41.5 / 45.6) Multiple recirculation pump trip (3.7/3.8)

QUESTION: 99 13406 (1 point(s))

Given the following conditions:

- ➢ Reactor power is 36%
- ➢ Recirculation pump B is idle
- ➢ Preparations to start "B" recirculation pump are in progress
- ➢ Bus 1C is supplied via the 1CS breaker
- ➢ The Start-up Transformer loses power

What action is required?

a. Manually scram the reactor.

b. Enter procedure 2.1.4, Normal Shutdown.

c. Enter procedure 2.1.4.1, Rapid Shutdown.

d. Close "B" Recirculation pump discharge valve, RR-MOV-MO53B.

ANSWER: 99 13406

a. Manually scram the reactor

Answer source: 2.4RR, p. 1, step 3.1

Distractors:

b. A scram is required.
c. A scram is required.
d. The "A" Recirc discharge valve must be shut, but not the "B".

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	12222	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	2	1	3	Multiple Choice	

Topic Area	Description
Administrative	RO only INT0320115, CHEMISTRY PROCEDURES

Related Lessons	
INT0320115	CHEMISTRY PROCEDURES

Related Objectives	
INT0320115E0E0200	Discuss the compliance and use requirements associated with RWP's.

Related References	
9.ALARA.4	Procedure 9.ALARA.4, Radiation Work Permits

Related Skills (K/A)	
2.3.2	Knowledge of facility ALARA program. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.5/2.9)

QUESTION: 100 12222 (1 point(s))

A worker was replacing pump motor coupling in the Radwaste Basement. The work was being performed under an SWP due to radiation levels of 120 mrem in the area of the pump. Shortly after commencing work on the pump coupling in the Radwaste Basement, the worker was immediately needed to assess an emergent problem with a reactor feed pump oil pump.

What are the MINIMUM actions needed in order to allow the worker to proceed to the reactor feed pump?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Log off the SWP.

b. Log onto the appropriate RCAWP.

c. Log off the SWP and log onto the appropriate RCAWP.

d. Log onto the appropriate RCAWP and log off of the SWP.

ANSWER: 100 12222

b. Log onto the appropriate RCAWP.

Logging onto the correct RCAWP automatically logs the worker off of the SWP.

Answer source: 9.ALARA.4, p. 9, step 7.7.2.2

Distractors:

a. The individual would need to log onto the appropriate RCAWP.
c. Because it is not necessary to perform the step of logging off the SWP.
d. As it is not necessary to log off the SWP.

Source: ***Direct from Bank***

Answer Key

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region: I / II / III / IV
Date: June 13, 2003	Facility/Unit: Cooper Nuclear Station
License Level: RO / SRO	Reactor Type: W / CE / BW / GE
Start Time: 08:00	Finish Time: 14:00

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected six hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	__________ Points
Applicant's Score	__________ Points
Applicant's Grade	__________ Percent

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
1	1111	02	03/28/2003	06/13/2003	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	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001130J	Predict the consequences of the following events on the AC Electrical Distribution System: Exceeding current limitations

Related References	
2.2.18	4160V Auxiliary Power Distribution System
2.2.20	Procedure 2.2.20, Standby AC Power System (Diesel Generator)

Related Skills (K/A)	
262001.A4.05	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Voltage, current, power, and frequency on A.C. buses (3.3/3.3)

QUESTION: 1 1111 (point(s))

Given the following conditions:

- ➢ Reactor is in Hot Shutdown
- ➢ DG 1 is paralleled to 1F for surveillance testing
- ➢ The startup transformer supply breaker 1AS trips
- ➢ DG 1 load reaches 150% of rated current

Which breaker(s) will trip?

a. **ONLY** EG1

b. **ONLY** 1AF

c. **BOTH** 1AF and 1FA

d. **BOTH** EG1 and 1FA

ANSWER: 1 1111

c. **BOTH** 1AF and 1FA

1FA is tripped by the over current condition 1AF trips because 1AF is in NORMAL AFTER CLOSE and Bus 1A is deenergized.

Answer source: 2.2.18, pp. 170, 171 (1AF), step 2.6.2, p. 173 (1FA), step 2.9.2

Distractors:

a. EG1 does not trip.
b. Both 1FA and 1AF will be tripped.
d. EG1 Remains closed to maintain 1F energized.

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
2	14036	00	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001070C	State the electrical power supplies to the following: PMIS Computer
COR0010102001080E	Predict the consequences of the following on plant operation: PMIS/UPS inverter failure
COR0010102001090C	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Automatic bus transfer
COR0010102001060D	Describe the interrelationship between the AC Electrical Distribution System and the following: PMIS/UPS

Related References	
2.2.63	Procedure 2.2.63, PMIS Uninterruptible Power Supply System

Related Skills (K/A)	
262002.K1.06	Knowledge of the physical connections and/or cause- effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Unit computer: Plant-Specific (2.6/2.7)

QUESTION: 2 14036 (1 point(s))

The plant was operating at power with the emergency transformer out of service. A fault occurred that resulted in the lockout of 4160V bus 1A and 4160V bus 1B. Both Diesel generators started and loaded their respective buses.

If MDP-2 were to deenergize at this time, what power, if any, would be immediately supplied to the PMIS computer?

PMIS would be . . .

a. deenergized.

b. powered directly from MDP-1.

c. powered from the 125 VDC PMIS battery via the inverter.

d. powered from MCC-L via the inverter and battery charger.

ANSWER: 2 14036

c. powered from the 125 VDC PMIS battery via the inverter.

The loss of the lockout experienced on the plant's busses resulted in the brief deenergization of MCC-L which results in a lockout of the feeder from MCC-L to PMIS for 15 minutes following reenergization. The PMIS 125VDC battery would assume the load via the inverter to power PMIS.

Answer source: 2.2.63, p. 10, step 1.2.1

Distractors:

a. PMIS would remain energized via the battery and inverter.
b. PMIS would remain energized via the battery and inverter, MDP-1 would automatically supply PMIS only if the inverter output failed.
d. MCC-L is locked out for 15 minutes following it's reenergization.

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
3	1099	01	03/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0010102, AC Electrical Distribution

Related Lessons
COR0010102 AC Electrical Distribution

Related Objectives	
COR0010102001090C	Describe the AC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Automatic bus transfer

Related References	
2.2.18	4160V Auxiliary Power Distribution System

Related Skills (K/A)	
295003.AA2.01	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.10 / 43.5 / 45.13) Cause of partial or complete loss of A.C. power. (3.4/3.7)

QUESTION: 3 1099 (1 point(s))

Given the following conditions:

- ➢ The reactor is shutdown
- ➢ 4160V bus 1F has just been transferred **from** the Emergency Transformer **to** Bus 1A (1FA has just been closed)
- ➢ DG1 is running unloaded
- ➢ Breaker 1FS control switch is still in the NORMAL AFTER CLOSE position

How will the electrical system respond to a loss of the Startup Transformer at this time?

a. Breaker 1FS will close **immediately**, regardless of how long Bus 1F has been de-energized.

b. Breaker 1FS will close 12.5 seconds after the loss of Bus 1F voltage occurred.

c. DG-1 will supply 4160V Bus 1F **immediately**, regardless of how long Bus 1F has been de-energized.

d. DG-1 will supply 4160V Bus 1F after the loss of voltage has existed on Bus 1F for at least 10 seconds.

ANSWER: 3 1099

d. DG-1 will supply 4160V Bus 1F after the loss of voltage has existed on Bus 1F for at least 10 seconds.

Due to 1FS being in the NORMAL AFTER CLOSE position it will not automatically reclose. Therefore the DG will be required to supply the bus. The DG breaker always waits at least 10 seconds with the loss of voltage relay energized before automatically closing.

Answer source: 2.2.20, p. 29, step 2.7.1.5

Distractors:

a,b Incorrect because 1FS is in the NORMAL AFTER CLOSE position.
c. Incorrect as the diesel always has an at least 10 second delay before it automatically will close onto bus 1F or 1G.

Source: *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
4	14035	00	03/31/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES

Related Lessons
COR0011302 OPS MAIN GENERATOR AND AUXILIARIES

Related Objectives	
COR0011302001060D	Describe the Main Generator and Auxiliaries design features and/or interlocks that provide for the following: Generator voltage regulation
COR0011302001080I	Predict the consequences of the following on the Main Generator and Auxiliaries: Grid instabilities
COR0011302001140D	Briefly explain the following concepts as they apply to the Main Generator: Reactive load

Related References	
2.2.14	22 KV Electrical System

Related Skills (K/A)	
245000.A4.14	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Generator megavar output (2.5/2.5)

QUESTION: 4 14035 (1 point(s))

The plant was operating at power with the following conditions present:

- ➢ Generator Load 765 Mwe
- ➢ Generator Reactive Load -55 MVAR
- ➢ Generator Voltage Regulator In Automatic control

Changing load conditions on the grid result in a slow increase in 345 KV voltage. Which of the following describes the **initial** effect of the changing voltage on main generator MVARs and field current?

MVARs become . . .

a. less negative and field amperage increases.

b. less negative and field amperage decreases.

c. more negative and field amperage increases.

d. more negative and field amperage decreases.

ANSWER: 4 14035

d. more negative and field amperage decreases.

With the increase in grid voltage and the voltage setpoint on the generator voltage regulator being held constant, the generator reactive load becomes more negative. The generator automatic voltage regulator will sense the increase in generator terminal voltage (due to the increase in 345 voltage) and reduces field amps to control voltage at the setpoint.

Answer source: 5.3GRID, p. 7, step 5.12, electrical theory

Distractors:

a. MVARs do not become less negative and field amperage does not increase.
b. MVARs do not become less negative.
c. Although MVARs do become more negative field amperage does not increase.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
5	14670	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0011402, OPS MAIN TURBINE

Related Lessons
COR0011402 OPS MAIN TURBINE

Related Objectives	
COR0011402001120A	Briefly describe the following concepts as they apply to Main Turbine and Auxiliaries: Feedwater heaters and Extraction Steam system operation

Related References	
2.2.29	Procedure 2.2.29, Feedwater Heaters And Extraction Steam System
2.2.77	Procedure 2.2.77, Turbine Generator

Related Skills (K/A)	
295005.AA2.03	Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.10 / 43.5 / 45.13) Turbine valve position (3.1/3.1)

QUESTION: 5 14670 (1 point(s))

The plant is operating at 90% power when the Main Generator trips.

Which of the following valves automatically **OPEN**?

a. Reheat stop valves.

b. Extraction steam dump valves.

c. Extraction Steam Non-Return valves.

d. Reactor Feed Pump Turbine low pressure steam supply valve.

ANSWER: 5 14670

b. Extraction steam dump valves.

Answer source: 2.2.29, p. 19, step 2.1

Distractors:

a, c, and d close as a result of a turbine trip.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
6	19124	01	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0011802, OPS Radiation Monitoring

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT COR0011802 OPS Radiation Monitoring

Related Objectives	
COR0022802001080A	Describe the Standby Gas Treatment design features and/or interlocks that provide for the following: Automatic system initiation
COR0022802001130A	Given plant conditions, determine if any of the following should occur: SGT automatic initiation
COR0011802001120E	Given plant conditions related to the Radiation Monitoring system, determine if any of the following should occur: Reactor Building Ventilation Isolation

Related References	
4.7.5	Procedure 4.7.5, Reactor Building Vent Exhaust Radiation Monitoring System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
295034.EK1.02	Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.8 to 41.10) Radiation releases (4.1/4.4*)

QUESTION: 6 19124 (1 point(s))

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

- ➢ RMP-RM-452A: 14 mrem/hr
- ➢ RMP-RM-452B: 7 mrem/hr
- ➢ RMP-RM-452C: 11 mrem/hr
- ➢ RMP-RM-452D: 13 mrem/hr

NO group isolations or automatic initiations occur.

What actions are required (if any) and why?
(Note: Use *actual* setpoints in your evaluation.)

a. **NO** actions are required because **neither** ***DIVISION*** logic has actuated.

b. **NO** actions are required because **only** the ***DIVISION I*** logic has actuated.

c. Manually start **only** "A" SGT train because **only** the ***DIVISION I*** logic has actuated.

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

ANSWER: 6 19124

d. Manually start **BOTH** SGT trains and isolate the Reactor Building ventilation because there is a start/isolation signal from **BOTH** Divisions.

If RMP-RM-452A or C AND RMP-RM-452B or D exceed 10 mrem/hr, Reactor Building isolates, and both SGT systems start. Per 2.0.3 "Operators shall validate automatic safety initiations and actuations. They shall ensure automatic actions take place in response to valid initiation signals"

Answer source: 4.7.5, pp. 5 & 6, steps 1.2.3, 1.2.4, 1.2.5, & 1.3.1.1

Distractors:

a,b,c Both Divisions should have actuated. The reactor building should have isolated and both SGT trains should have started.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
7	5084	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0012002, OPS Reactor Water Cleanup

Related Lessons
COR0012002 OPS Reactor Water Cleanup

Related Objectives	
COR0012002001090D	Describe the RWCU design features and/or interlocks that provide for the following: Piping over-pressurization protection
COR0012002001130G	Given a RWCU component manipulation, predict and explain the changes in the following parameters: RWCU system pressure

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
2.1.31	Ability to locate control room switches / controls and indications and to determine that they are reflecting the desired plant lineup. (CFR: 45.12) (4.2/3.9)

QUESTION: 7 5084 (1 point(s))

The plant was operating at power when a RWCU isolation (Group 3) occurred.

What change in RWCU system lineup is designed to prevent overpressurization of Reactor Water Cleanup (RWCU) System Piping?

a. Return Isolation Valve, MO-68 is cracked open.

b. Blowdown Flow Control Valve PCV-55 is closed.

c. Demin Suction Bypass Valve MO-74 is cracked open.

d. Drain Valve to Radwaste System MO-57 and Drain Valve to the Condenser MO-56 are both cracked open.

ANSWER: 7 5084

c. Demin Suction Bypass Valve MO-74 is cracked open.

Following a RWCU isolation Procedure 2.1.22 requires that MO-74 be cracked open to prevent overpressurization by mini-purge. CRD purge of RWCU Pump seals can overpressurize the pump and piping following closure of MO-15 or MO-18. Opening MO-74 provides a path for CRD flow around the demins to the Reactor Vessel.

Answer source: 2.1.22, p. 10, step 6.4

Distractors:

a. MO-68 should already be open and this valve alone would not provide overpressure protection from mini-purge following isolation because a path around the now out of service demineralizers is required.
b. This valve should already be closed, in addition its closure would do nothing to prevent overpressurization of the RWCU piping. FCV-55 closes to protect downstream piping from high pressure or upstream piping from low pressure.
d. These valves should not be opened simultaneously as this could result in a loss of vacuum.

Source: ***Modified from 5084***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
8	14043	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Lessons
COR0012402 OPS TURBINE EQUIPMENT COOLING SYSTEM

Related Objectives	
COR0012402001020D	Describe the interrelationships between the TEC system and the following: Control Room HVAC

Related References	
2.2.76	Procedure 2.2.76, Turbine Equipment Cooling Water System

Related Skills (K/A)	
290003.K1.05	Knowledge of the physical connections and/or cause- effect relationships between CONTROL ROOM HVAC and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Component cooling water systems (2.8/3.0)

QUESTION: 8 14043 (1 point(s))

What provides the normal and the backup cooling water to the Control Room Air conditioner?

The normal supply is from the . . .

a. Turbine Equipment Cooling (TEC) system and the backup supply is from the Service Water (SW) system.

b. Reactor Equipment Cooling (REC) system and the backup supply is from the Service Water (SW) system.

c. Turbine Equipment Cooling (TEC) system and the backup supply is from the Reactor Equipment Cooling (REC) system.

d. Reactor Equipment Cooling (REC) system and the backup supply is from the Turbine Equipment Cooling (TEC) system.

ANSWER: 8 14043

a. Turbine Equipment Cooling (TEC) system and the backup supply is from the Service Water (SW) system.

TEC supplies cooling to the Control Room Air Conditioner and can be supplied from SW by manually positioning local valves.

Answer source: 2.2.76, p. 33, step 1.2.2.7.

Distractors:

b. REC is not capable of providing the normal supply the Control Room AC unit.
c. REC is not capable of supplying the backup cooling to the Contol Room AC unit.
d. REC is not capable of providing the normal supply the Control Room AC unit and TEC is the normal supply.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
9	14045	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0013001, HWC Gas Generation System

Related Lessons
COR0013001 HWC Gas Generation System

Related Objectives	
COR0013001001040D	Identify the reason/function of the following systems interface and general physical location of the interface with the HWC system: Offgas System

Related References	
2.2.98	Hydrogen/Oxygen Generation System

Related Skills (K/A)	
272000.K5.01	Knowledge of the operational implications of the following concepts as they apply to RADIATION MONITORING SYSTEM: (CFR: 41.7 / 45.4) Hydrogen injection operation's effect on process radiation indications: Plant-Specific (3.2/3.5)

QUESTION: 9 14045 (1 point(s))

The plant is operating at 100% power with the hydrogen injection in service when OWC INJECTION SYS SHUTDOWN, A-3/F-4 alarms. The Control Room operator places the OWC INJECTION SYS ENABLE SWITCH to SHUTDOWN and verifies the that the green (Shutdown) light is on.

How does this affect ERP radioactive release rate and Main Steam Line (MSL) radiation level?

ERP release rate . . .

a. increases and MSL radiation levels increase.

b. decreases and MSL radiation level decrease.

c. is unchanged and MSL radiation level decrease.

d. is unchanged and MSL radiation level is unchanged.

ANSWER: 9 14045

c. is unchanged and MSL radiation level decrease.

The indications, annunciator and operator action indicate a loss of hydrogen injection. The loss of the hydrogen injection results in a shift of the ratio of N-16 as ammonia or ammoniun to nitrate or nitrite anion forms. This results in less carryover of N-16 out the main steam lines and a reduction in MSL radiation levels. Since N-16 has a short half life this change in carryover does not effect the release rate out the ERP.

Answer source: COR012-03-01, p. 4

"What is going to happen if Cooper starts adding hydrogen to the reactor water? According to the previous paragraph, if the nitrogen reacts with hydrogen, ammonia if formed. With a lot more hydrogen in the Reactor to combine with, the nitrogen will combine with it. Consequently there will be a lot more nitrogen-16 going over to the turbine and dose rates will be much higher."

Distractors:

a. ERP release rate does not increase.
b. ERP release rate does not decrease.
d. MSL radiation levels decrease.

<u>Source:</u> ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
10	2931	02	03/18/2003	06/13/2003	Licensed Operator	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001050F	Describe the interrelationship between the Primary Containment system and the following: Plant Air
COR0020302001120F	Describe the Containment design features and/or interlocks that provide for the following: Reactor building to Torus D/P

Related References	
3.6.1.7	Reactor building-to-suppression chamber vacuum breakers
COR0020302	Containment

Related Skills (K/A)	
295019.AK2.09	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: (CFR: 41.7 / 45.8) Containment. (3.3/3.3)

QUESTION: 10 2931 (1 point(s))

Drywell sprays are in service to support EOP actions when a complete loss of instrument air (including nitrogen) occurs. Due to a logic failure, the drywell spray valves (RHR-MO-26A and RHR-MO-31A) have been opened manually using the local handwheels.

Is the torus protected from exceeding design negative pressure under these conditions and why/why not?

a. Yes, all reactor building-to-torus vacuum breakers are motor-operated.

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

c. No, the reactor building-to-torus vacuum breakers fail closed on a loss of air.

d. No, the reactor building-to-torus vacuum breakers are not designed to facilitate this amount of flow.

ANSWER: 10 2931

b. Yes, the reactor building-to-torus vacuum breakers fail in such a manner as to prevent an excessive negative pressure in the torus.

Answer source: COR002-03-02, p. 20

Distractors:

a. One of the vacuum breakers is pneumatically operated.
c. The MOV doesn't fail anywhere on loss of air. The AOV fails open.
d. The vacuum breakers are sized to facilitate this flow.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
11	10081	02	03/21/2003	06/13/2003	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	COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001080D	State the electrical power supplies to the following: N2 solenoid valve
COR0020302001240B	Predict the consequences of a malfunction of the following on PCIS: DC electrical.
COR0020302001230C	Predict the consequences of a malfunction of the following on the Primary containment: Containment atmospheric control/nitrogen make-up.

Related References	
2.2.60	Procedure 2.2.60, Primary Containment Cooling And Nitrogen Inerting System
2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1
2.2.59	Procedure 2.2.59, Plant Air System

Related Skills (K/A)	
2.1.23	Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) (3.9/4.0)

QUESTION: 11 10081 (1 point(s))

The plant is at 100% power with the following conditions:

- ➢ Annuciator 9-3-1/G-2, NITROGEN SOLENOID DE-ENERGIZED is in alarm
- ➢ 125 VDC Panel AA2 is de-energized

Which of the following would be the ***quickest*** action that will restore pressure to the drywell pneumatic header?
(NOTE: The choices are listed from QUICKEST to LONGEST order.)

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

b. Open the Reactor building drywell supply air valve (IA-V-571) above the Southeast Hydraulic Control Units.

c. Open RR-SPV-740 AND RR-SPV-741 SUPPLY SHUTOFF (IA-1672) near RWCU precoat pump.

d. Hook up the nitrogen bottles that are stored in a rack near the header.

ANSWER: 11 10081

a. Open the cross-connect valve (IA-SOV-SPV21) from instrument air to the drywell pneumatic header using a switch on panel 9-3.

Answer source: 2.3_9-3-1, p. 74,
2.2.59 p. 25, step 1.2.11

Distractors:

b. This would take a person some time to manually open the valve (faster than bottles).
c. These valves are already open and would not restore drywell pneumatics.
d. This would take at least one person, some tools and time.

2.1.23 Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45: 45.2 / 45.13) as it applies to: 223002 PCIS/Nuclear Steam Supply Shutoff

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
12	5155	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020402, CONTROL ROD DRIVE HYDRAULICS

Related Lessons
COR0020402 CONTROL ROD DRIVE HYDRAULICS

Related Objectives	
COR0020402001140A	State the electrical power supply to the following CRDH components: CRDH pumps motors.

Related References	
5.3EMPWR	EMERGENCY POWER
2.2.8	Procedure 2.2.8, Control Rod Drive System
2.2.8A	Procedure 2.2.8A, Control Rod Drive Hydraulic System Valve Checklist

Related Skills (K/A)	
201001.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: (CFR: 41.7 / 45.7) A.C. power (3.3/3.3)

QUESTION: 12 5155 (1 point(s))

Given the following conditions:

- The plant is operating at 65% power
- The "B" Control Rod Drive (CRD) Pump is running
- 480 VAC Critical Switchgear Bus 1G trips

What is the status of "B" CRD pump and Drive Water DP?

a. The "B" Pump is running.
Drive Water DP is unaffected.

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

c. The "B" Pump is stopped.
Drive Water DP will decay away over the next several minutes.

d. The "B" Pump is running.
The Drive Header Pressure Control Valve has lost power.

ANSWER: 12 5155

b. The "B" Pump is stopped.
Drive Water DP will rapidly lower to zero.

480 VAC Bus 1G provides power to CRD Pump "B" so Pump "B" is stopped. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.

Answer source: 2.2.8A, p. 10

Distractors:

a. The "B" pump is powered by 480 VAC Bus 1G and will trip.
b. With no pump flow, drive header pressure will rapidly lower to Reactor pressure due to flow to the cooling header and Ref Leg Fill. Further reduction will be more gradual due to some check valve leakage, but Drive Water DP will quickly lower, as Drive pressure, to zero.
d. The "B" pump is powered by 480 VAC Bus 1G and will trip.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
13	14040	00	03/13/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020602, CORE SPRAY

Related Lessons
COR0020602 CORE SPRAY

Related Objectives	
COR0020602001080H	Given a Core Spray component manipulation, predict and explain the changes in the following: System lineup
COR0020602001120A	Given plant conditions, determine if any of the following Core Spray Actions should occur: System initiation.
COR0020602001120D	Given plant conditions, determine if any of the following Core Spray Actions should occur: Valve reposition.
COR0020602001050E	Describe the Core Spray system design features and/or interlocks that provide for the following: Pump minimum flow

Related References	
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
209001.A3.03	Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: (CFR: 41.7 / 45.7) System pressure (3.5/3.5)

QUESTION: 13 14040 (1 point(s))

The plant was operating at power with the "A" Core Spray subsystem in full flow test at 5000 gpm. An accident occurs that results in increasing drywell pressure and lowering reactor water level and lowering reactor pressure. The following plant conditions exist:

- ➢ Drywell pressure 11 psig (rising)
- ➢ Reactor water level -21" (wide range, lowering)
- ➢ Reactor pressure 375 psig (lowering)

What is the pressure response of the A Core Spray system ***at this time***?

Core spray system <u>pressure</u> . . .

a. remains the same.

b. increases to pump shut-off head.

c. decreases to <u>just above</u> reactor pressure.

d. increases to <u>just below</u> pump shut-off head.

ANSWER: 13 14040

d. increases to <u>just below</u> pump shut-off head.

An initiation signal is present for the Core Spray System. The CS test valve would receive a close signal resulting in a significant reduction in flow and since reactor pressure remains above the shut-off head for the pumps flow would be reduced to the point that the minimum flow valve would open. Core Spray system pressure would then be just below pump shut-off head.

<u>Answer source:</u> 2.2.9 p. 20

Distractors:

a. Core Spray pressure would not remain the same because system flow rate would become significantly reduced when the initiation signal occurred and the test line isolated.
b. Core Spray system pressure would be below shut-off head because of the flow that exists through the minimum flow valve.
c. Core Spray pressure increases.

<u>Source:</u> *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
14	14050	00	03/28/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020702, OPS DC ELECTRICAL DISTRIBUTION

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION

Related Objectives	
COR0020702001090A	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Manual/automatic transfers of control
COR0020702001090B	Describe the DC Electrical Distribution System design feature(s) and/or interlock(s) that provide for the following: Breaker interlocks, permissives, bypasses and crossties

Related References	
2.2.25.2	125 VDC ELECTRICAL SYSTEM (DIV 2)

Related Skills (K/A)	
263000.K4.02	Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Breaker interlocks, permissives, bypasses and cross ties: Plant-Specific (3.1/3.5)

QUESTION: 14 14050 (1 point(s))

The plant is operating at rated power when the normal power supply to the 125VDC HPCI Starter Rack is lost.

What interlocks exist for transfer of the HPCI Starter Rack to its alternate supply?

Inadvertent transfer of the 125VDC HPCI Starter Rack from its normal to its alternate supply is prevented by transfer switch design . . .

a. **only**.

b. **AND** the alternate supply breaker is locked open **only**.

c. **AND** a mechanical interlock prevents closing both supply breakers simultaneously **only**.

d. **AND** the alternate supply breaker is locked open **AND** a mechanical interlock prevents closing both supply breakers simultaneously.

ANSWER: 14 14050

b. **AND** the alternate supply breaker is locked open **only**.

Answer source: 2.2.25.2 pp. 35 & 36, sections 38.3, 38.4, 38.5 & 38.6

Distractors:

a. The alternate supply is locked open.
c. The alternate supply is locked open and both switches are closed at the same time during a transfer.
d. Both switches are closed at the same time during a transfer.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
15	19090	01	03/28/2003	06/13/2003	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
Integrated Plant	COR0020702, OPS DC ELECTRICAL DISTRIBUTION

Related Lessons
COR0020702 OPS DC ELECTRICAL DISTRIBUTION COR0021202 INTERMEDIATE RANGE MONITOR

Related Objectives	
COR0020702001080L	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: SRMs
COR0020702001080R	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Radiation Monitoring systems
COR0021202001070B	Predict the consequences of a loss or malfunction of the following would have on the IRM system: 24/48 VDC
COR0020702001080J	Given a specific DC Electrical Distribution system malfunction, determine the effect on any of the following: Reactor Protection system

Related References	
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.2.26	Procedure 2.2.26, 24 VDC Electrical System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
215003.A3.03	Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7 / 45.7) RPS status (3.7/3.6)

QUESTION: 15 19090 (1 point(s))

The station is in MODE 2 withdrawing control rods in an approach to criticality during a startup. The following equipment simultaneously trips:
(NOTE: Other equipment also trips but is not required to assess conditions.)

- IRM "A", "C", "E" and "G"
- SRM "A" and "C"
- Off-Gas Radiation monitor "A"
- Reactor Building Vent Radiation monitors "A" and "C"
- Control rods remain at their pre-transient position
- **NO** group isolations have occurred

What occurred and what actions (if any) are required?

a. A loss of RPSPP "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

b. A loss of RPSPP "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

c. A loss of 24 VDC "A" has occurred. Manually initiate a Reactor scram and a Group 6 isolation.

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

ANSWER: 15 19090

d. A loss of 24 VDC "A" has occurred. **NO** operator actions are required to compensate for logic actuation failure.

The loss of 24 vdc will cause all these instruments to become inoperative. No scram or group isolation will occur due to this single power loss.

<u>Answer source:</u> 2.2.22, p. 9 (RPS loss distractors),
2.2.26, step 2.2.1

Distractors:

a. RPS power loss would not cause the loss of IRMs/SRMs. No ATWS or group isolation failure has occurred.
b. RPS power loss would not cause the loss of IRMs/SRMs.

c. No ATWS or group isolation failure has occurred.

Source: ***Direct From Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
16	1507	02	03/31/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020802, Diesel Generators

Related Lessons
COR0020802 DIESEL GENERATORS COR0010102 AC Electrical Distribution

Related Objectives	
COR0020802001090E	Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Load Shedding and Sequencing
COR0010102001130B	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of coolant accident
COR0010102001130C	Predict the consequences of the following events on the AC Electrical Distribution System: Loss of off-site power

Related References	
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation
COR0020802	Diesel Generators

Related Skills (K/A)	
264000.K5.06	Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3) Load sequencing (3.4/3.5)

QUESTION: 16 1507 (1 point(s))

The plant was operating at rated power when a loss of all off-site power occurred coincident with a large recirculation suction line break.

What will be the sequential loading of emergency buses?
(T = DG Output Breaker Closure)

a. T+0 the Core Spray pump starst;
T+5 seconds the first RHR Pump starts;
T+10 seconds the second RHR pump and SGT start.

b. T+0 the first RHR pump starts;
T+5 seconds the Core Spray pump and SGT start;
T+10 seconds the second RHR pump starts.

c. T+0 the first RHR pump and SGT starts;
T+5 seconds the Core Spray pump starts;
T+10 seconds the second RHR pump starts.

d. T+0 the first RHR pump and SGT starts;
T+5 seconds the second RHR pump starts;
T+10 seconds the Core Spray pump starts.

ANSWER: 16 1507

d. T+0 the first RHR pump and SGT starts;
T+5 seconds the second RHR pump starts;
T+10 seconds the Core Spray pump starts.

Answer source: COR002-08-02, p. 65, Table 1

Distractors:

a. RHR pump starts first and second, CS starts last. SGT starts at T=0.
b. RHR pump starts first and second, CS starts last. SGT starts at T=0.
c. RHR pump starts first and second, CS starts last.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
17	14032	00	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021102, OPS High Pressure Coolant Injection (HPCI)

Related Lessons
COR0021102 OPS High Pressure Coolant Injection (HPCI)

Related Objectives	
COR0021102001080N	Describe the HPCI design features and/or interlocks that provide for following: Pump minimum flow
COR0021102001100H	Predict the consequences of the following on the HPCI system: Low ECST level
COR0021102001080M	Describe the HPCI design features and/or interlocks that provide for following: Protection against draining the CST to the torus

Related References	
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
206000.A4.07	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Condensate storage tank level: BWR-2,3,4 (3.5/3.5)

QUESTION: 17 14032 (1 point(s))

The plant was operating at power with the HPCI system in full flow test per 6.HPCI.103 when annunciator 9-3-2/A-4, HPCI SUCTION TRANSFER alarms. ECST level is 22".

What is the status/alignment of HPCI several minutes later?
(Assume the operator takes no action.)

HPCI suction valves are aligned to the suppression pool, the Minimum Flow Valve (MO-25) is _________ and the Pump Test Return Line Isolation Valves (MO-21 & 24) are _________.

a. open closed

b. closed closed

c. open open

d. closed open

ANSWER: 17 14032

a. open closed

The low ECST level has initiated a swap of the of the HPCI suction valves. With the swap over to suction from the suppression pool the pump test return isolation valves automatically close. Now the HPCI system is without a discharge path the minimum flow valve opens due to low flow.

Answer source: 2.2.33 p. 15, steps 2.1.1.8 (MO-25), 2.1.1.9 (MO-21) & 2.1.1.10 (MO-24)

Distractors:

b. The minimum flow valve is open.
c. The Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.
d. The minimum flow valve is open and the Pump Test Return Line Isolation Valves, (MO-21 & 24) are closed.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
18	19051	01	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0021102, OPS High Pressure Coolant Injection (HPCI)

Related Lessons
COR0021102 OPS High Pressure Coolant Injection (HPCI) SKL0124211 HIGH PRESSURE COOLANT INJECTION

Related Objectives	
COR0021102001120A	Given plant conditions, determine if the following HPCI actions should occur: System initiation
SKL012421100A030J	Given plant conditions, predict changes in the following HPCI system components/parameters: Turbine speed
SKL012421100B0600	Comply with all related HPCI system limits and precautions.
COR0021102001100V	Predict the consequences of the following on the HPCI system: High reactor water level
SKL012421100A0200	Explain the HPCI system limitations and precautions as stated in the SOP 2.2.33 and SOP 2.2.33.1.

Related References	
791E271	HPCI System Elementary Diagram
2.2.33	Procedure 2.2.33, High Pressure Coolant Injection System

Related Skills (K/A)	
2.1.32	Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) (3.4/3.8)

QUESTION: 18 19051 (1 point(s))

The plant was operating at power when a loss of off-site power occurred. The reactor scrammed and HPCI started on low reactor water level. Reactor water level quickly recovered and the HPCI turbine tripped on high RPV water level. The following plant conditions were present:

- ➢ Reactor water level 45" (NR) (lowering slowly)
- ➢ Reactor pressure 850 psig (rising slowly)
- ➢ Drywell pressure 2.2 psig (rising slowly)

What is/are the **MINIMUM** action(s) required to restart HPCI ***at this time***?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

b. Momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

c. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

d. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **AND** momentarily depress the Initiation Signal Reset pushbutton **AND** place the HPCI-MO-14 control switch to open.

ANSWER: 18 19051

a. Momentarily depress the Reactor Hi Water Level Signal Reset pushbutton **ONLY**.

Per 2.2.33.1:

CAUTION - If HPCI initiation signal cannot be reset, HPCI System will automatically start when RX HI WTR LEVEL SIGNAL RESET pushbutton is depressed and vessel level is $\leq$ +54".

During the transient drywell pressure has risen to greater than the initiation setpoint for HPCI. Since an automatic initiation signal is present, if the operator depresses the Reactor Hi Water Level Signal Reset pushbutton the system will reinitiate.

Answer source: 2.2.33.1, p. 8
2.2.33, p. 20, step 2.2.6

Distractors
b, c, d - only the high level trip reset need be depressed.

2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12) as it applies to: 295008 High Reactor Water Level / 2

<u>Source:</u> ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
19	16513	01	02/27/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021402, Digital Electro-Hydraulic Control

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040B	Describe how the DEH control system operates to control the following: Reactor pressure
COR0020902001070B	Given a specific DEH Control system malfunction, determine the effect on any of the following: Reactor pressure

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

Related Skills (K/A)	
241000.K4.01	Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature (s) and/or interlocks which provide for the following: (CFR: 41.7) Reactor pressure control (3.8/3.8)

QUESTION: 19 16513 (1 point(s))

The plant is operating at 100% power when the in-service DEH pressure controller fails such that controller output INCREASES slowly.

Assuming NO operator action is taken, what is the plant response?

a. The reactor scram due to high reactor pressure.

b. The MSIVs isolate due to low reactor pressure.

c. Turbine throttle pressure will be controlled about 4 psig LOWER than before the failure.

d. Turbine throttle pressure will be controlled about 4 psig HIGHER than before the failure.

ANSWER: 19 16513

b. The MSIVs isolate due to low reactor pressure.

As the controller output increases the turbine governor valves would open in response to the controller output. The opening of the valves would reduce reactor pressure and result in a MSIV closure due to low MSL pressure with the mode switch in RUN.

Answer source: COR002-09-02, p. 49, section "c."

Distractors:

a: Reactor pressure will lower as controller output signals the TCVs to OPEN.
c: The backup pressure regulator is set for a pressure 4 psi higher.
d: Reactor pressure will lower as controller output signals the TCVs to OPEN.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
20	1058	01	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021502, NUCLEAR BOILER INSTRUMENTATION

Related Lessons
COR0021502 NUCLEAR BOILER INSTRUMENTATION

Related Objectives	
COR0021502001040H	Briefly describe the following concepts as they apply to NBI: Recirculation flow effects on level indicators

Related References	
COR0021502	Nuclear Boiler Instrumentation

Related Skills (K/A)	
295001.AA1.07	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: (CFR: 41.7 / 45.6) Nuclear boiler instrumentation system. (3.1/3.2)

QUESTION: 20 1058 (1 point(s))

The plant is operating at 100% power when both Reactor Recirculation pumps slowly run back to minimum speed. No operator action is taken.

What is the difference between ACTUAL and INDICATED Wide Range Reactor Water Level ***prior*** to the power reduction AND what is the expected change in that difference ***during*** the power reduction?

Prior to the power reduction, *actual* downcomer level is ___________ than *indicated* downcomer level AND the difference will get ___________ during the power reduction.

a. lower larger

b. higher larger

c. lower smaller

d. higher smaller

ANSWER: 20 1058

d. higher smaller

Due to the velocity affects of flow in the annulus, the variable leg will sense a lower pressure than is exerted by the height of water alone. This is seen as a lower indicated level. At higher recirc flows, higher velocities cause a greater difference between Wide Range indicated and actual levels. The difference between indicated and actual levels can range from 4-18"

Answer source: COR002-15-02, p. 42 & 43, section 2.c, d, f, g, & h.

Distractors:

a. Actual level is higher than indicated level. The difference will get smaller as power is reduced.
b. The difference will get smaller as power is reduced.
c. Actual level is higher than indicated level.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
21	5425	01	03/24/2003	06/13/2003	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	COR0021602, OPS NUCLEAR PRESSURE RELIEF

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001050J	Describe the Nuclear Pressure Relief system design features and/or interlocks that provide for the following: Safety/Relief operating signals
COR0021602001030J	Describe the interrelationships between the Nuclear Pressure Relief system and the following: RPS (low-low set initiation)

Related References	
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
239002.K6.03	Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: (CFR: 41.7 / 45.7) A.C. power: Plant-Specific (2.7*/2.9*)

QUESTION: 21 5425 (1 point(s))

Given the following:

- ➢ The plant is operating at 100% power at Beginning of Life (BOL)
- ➢ A loss of off-site power occurs
- ➢ All control rods fully inserted
- ➢ Pressure rises to 1090 psig, **THEN** lowers to 875 psig
- ➢ 20 minutes later, pressure is cycling between 1015 **AND** 875 psig

What is the status of Low Low Set (LLS) and why?
(NOTE: **NO** operator action is taken.)

a. LLS is controlling pressure. LLS logic has no AC powered inputs or components.

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

c. LLS is **NOT** controlling pressure. With RPS power unavailable, the SRVs must operate on mechanical relief setpoint to control pressure.

d. LLS is **NOT** controlling pressure. A fault must exist in the LLS logic as all conditions are present for LLS to automatically control pressure.

ANSWER: 21 5425

b. LLS is controlling pressure. With RPS power unavailable, the LLS logic can ***arm*** irrespective of reactor pressure.

Answer source: COR002-16-02, p. 24 & p. 24, p. 43 section E,
COR002-16-02 Figure 6

Distractors:

a. LLS logic is armed by RPS high pressure signal.
c. LLS logic will arm on high pressure with no RPS power available.
d. LLS is controlling pressure.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
22	5608	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021602, OPS NUCLEAR PRESSURE RELIEF

Related Lessons
COR0021602 OPS NUCLEAR PRESSURE RELIEF

Related Objectives	
COR0021602001020A	State the electrical power supply to the following NPR components: ADS logic
COR0021602001080F	Predict the consequences a malfunction of the following would have on the NPR system: D.C. power

Related References	
791E253	Automatic Blowdown System
2.2.1	Automatic Depressurization System

Related Skills (K/A)	
218000.K2.01	Knowledge of electrical power supplies to the following: (CFR: 41.7) ADS logic (3.1*/3.3*)

QUESTION: 22 5608 (1 point(s))

An accident has occurred, resulting in the following conditions:

➢	Reactor pressure	720 psig (lowering)
➢	RPV water level	-120" (WR stable)
➢	Drywell pressure	6.2 psig (rising)
➢	125 VDC panel AA2	De-energized

If present conditions continue, how will ADS respond?

ADS valves will . . .

a. ***fail to open*** due to loss of logic power.

b. ***fail to open*** due to RPV water level conditions not met.

c. be opened by the B logic circuit powered from its ***normal*** power source.

d. be opened by both logic circuits powered from their ***alternate*** power sources.

ANSWER: 22 5608

c. be opened by the B logic circuit powered from its ***normal*** power source.

Answer source: COR002-16-02, p. 21, & p. 22, section 3, p. 41
COR002-16-02 Figures 4 & 5

Distractors:

a. ADS will initiate powered from BB2.
b. ADS will initiate.
d. ADS "A" has no alternate source and is de-energized.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
23	14679	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM

Related Lessons
COR0021702 PLANT MANAGEMENT INFORMATION SYSTEM

Related Objectives	
COR0021702001070A	Given a specific PMIS malfunction, determine the effect on any of the following: On Demand print out.

Related References	
2.6.3	Procedure 2.6.3, COMPUTER SYSTEMS OPERATION AND OUTAGE RECOVERY
2.4COMP	Computer Malfunction

Related Skills (K/A)	
2.1.19	Ability to use plant computer to obtain and evaluate parametric information on system or component status. (CFR: 45.12) (3.0/3.0)

QUESTION: 23 14679 (1 point(s))

A plant shutdown is in progress with reactor power at 30% of rated and both PMIS computers in service when a loss of the primary PMIS computer occurs.

What is the impact of these conditions on plant operation?

Official Cases are . . .

a. available and, if the shutdown were to continue, RWM would be available.

b. available and, if the shutdown were to continue, RWM would be **UN**available.

c. **UN**available and, if the shutdown were to continue, RWM would be available.

d. **UN**available and, if the shutdown were to continue, RWM would be **UN**available.

ANSWER: 23 14679

a. available and, if the shutdown were to continue, RWM would be available.

When both PMIS computers are unavailable, monitoring functions (Computer Edits) 3D Monicore Official Cases, process parameter alarm monitoring and many other important functions are lost; however, in this instance, with the backup computer available, a Loss of primary computer will result in automatic fail-over to the backup computer, so there is no immediate impact on plant operation, all computer functions are available.

Answer source: COR002-17-02, p. 9, section B.1

Distractors:

b. The backup computer is available so the RWM remains available on the subsequent shutdown.
c. The backup computer is available so computer on-demand printouts remain available.
d. The backup computer is available so computer on-demand printouts remain available as does the RWM on the susequent shutdown.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
24	14451	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)

Related Lessons
COR0021802 OPS Reactor Core Isolation Cooling (RCIC)

Related Objectives	
COR0021802001120D	Given plant conditions, determine if the following RCIC actions should occur: Minimum flow valve position change
COR0021802001120A	Given plant conditions, determine if the following RCIC actions should occur: RCIC system initiation
COR0021802001100O	Predict the consequences of the following on the RCIC system: RCIC Turbine control system failure
COR0021802001120E	Given plant conditions, determine if the following RCIC actions should occur: RCIC turbine trip

Related References	
2.2.67	Procedure 2.2.67, Reactor Core Isolation Cooling System

Related Skills (K/A)	
217000.K5.02	Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): (CFR: 41.5 / 45.3) Flow indication (3.1/3.1)

QUESTION: 24 14451 (1 point(s))

The Reactor Core Isolation Cooling (RCIC) flow transmitter has failed low such that it senses 0 gpm irrespective of actual RCIC flow.

What is the expected RCIC system response upon receipt of a valid initiation signal?

The RCIC turbine will start and . . .

a. run normally.

b. trip on overspeed.

c. run continuously at minimum speed.

d. run continuously at approximately 4500 rpm.

ANSWER: 24 14451

d. run continuously at approximately 4500 rpm.

Loss of flow signal input to the flow controller results in a maximum speed demand signal. Since the output of the control box is limited to 50 milliamps, the turbine speed will top out at approximately 4500 rpm.

<u>Answer source:</u> COR002-18-02, p. 56, section 2

Distractors:

a. RCIC will not run normally.
b. The ramp generator is still functional on startup. RCIC RPM will not exceed 4500 when controller output is at 100%. Overspeed occurs at 5625 RPM.
c. RCIC will not run at mimimum speed.

<u>Source:</u> *Direct from bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
25	14027	00	02/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Generic Fundamentals	ACD0070307, Thermal Hydraulics (GP)

Related Lessons
ACD0070306 Heat Transfer & Heat Exchangers (GP) ACD0070307 Thermal Hydraulics (GP)

Related Objectives	
ACD00703020010800	Apply saturated and superheated steam tables in solving liquid-vapor problems.
ACD00703060010500	Solve heat flux and heat transfer rate problems.
ACD0060507001310C	Explain the relationship between decay heat generation and: time since reactor shutdown

Related References	
ACD0060507	Reactor Operational Physics

Related Skills (K/A)	
295007.AK1.02	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: (CFR: 41.8 to 41.10) Decay heat generation (3.1/3.4)

QUESTION: 25 14027 (1 point(s))

The plant has been shutdown for several days following a long operating cycle. Shutdown Cooling has been placed in service and a cooldown established. The following conditions are present:

- Reactor pressure is 15 psig
- Reactor vessel inventory is 350,000 lbm
- Decay heat rate is 0.6% of rated thermal power
- Ambient heat loss is 7.5 MWt
- Reactor coolant specific heat capacity is 1.08 BTU/lbm°F

A station blackout then occurs.

What reactor pressure will exist **two** (2) hours following this loss of all AC power?
(Assume that reactor inventory and ambient losses remain constant for the entire two hours.)

a. 83 psig

b. 90 psig

c. 165 psig

d. 180 psig

ANSWER: 25 14027

c. 165 psig

The thermal power of the reactor is 14.3 Mwt (.006*2381=14.3Mwt). The thermal power that is absorbed in the coolant is 14.3Mwt - 7.5 Mwt (ambient loss) =6.8Mwt. 6.8 Mwt is converted to BTU/hr by multiplying by 3.41E6. This yields 2.31E7 BTU/hr. Since the question asks for the conditions 2 hours after the loss of AC power this heat rate continues for 2 hours so 2.31E7BTU/hr is multiplied by 2 hrs to yield the total BTUs absorbed by the coolant or 4.62E7 BTUs. Now the known values can be substituted into the following equation to solve for the final temperature. $Q=Mc_p\Delta T$ The final temperature is calculated to be 373°F. Now steam tables are used to find the saturation pressure for that temperature (180 psia). The value is then converted to psig and the final answer of 162 psig. As the RPV is an enclosed volume under these conditions, addition of energy to the mass of water will result in a temperature change and very little phase change. The amount of energy lost to the latent heat of vaporization is insignificant when compared to the total enthalpy of the water in the RPV.

Answer source: Steam Tables, Generic Fundamentals

Distractors:

a. Would only be obtained if the candidate failed to account for two hours after the loss of AC power.
b. Would be obtained if the candidate failed to account two hours since the loss of AC power and failed to convert that answer to psig.
d. Would be only be obtained if the candidate failed to convert the final answer to psig.

<u>Source</u>: ***New***

<u>Provide to Candidate:</u> Calculator, Steam Tables, GFES Formula sheet.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
26	2521	03	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021902, REACTOR EQUIPMENT COOLING

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001060K	Given a specific REC malfunction, determine the efect on any of the following: Fuel Pool Cooling system

Related References	
2.2.65	Procedure 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: 26 2521 (1 point(s))

Closure of which of the following REC valves could lead to an increase in the fuel pool water temperature and in increase in airborne contamination?

a. Drywell Supply Isolation (REC-MO-702).

b. Non-Critical Header Supply (REC-MO-700).

c. Augmented Radwaste Supply (REC-MO-1329).

d. Critical Loop Return Crossover Valve (REC-MO-694).

ANSWER: 26 2521

b. Non-Critical Header Supply (REC-MO-700).

Answer source: 2.2.65, pp. 12 & 13, section 1.2

Distractors:

a. FPC is supplied by the non-critical header.
c. FPC is supplied by the non-critical header.
d. FPC is supplied by the non-critical header.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
27	14039	00	03/11/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0021902, REACTOR EQUIPMENT COOLING

Related Lessons
COR0021902 REACTOR EQUIPMENT COOLING

Related Objectives	
COR0021902001110A	Given plant conditions, determine if any of the following should occur: Non-Critical loop isolation
COR0021902001110C	Given plant conditions, determine if any of the following should occur: Any REC valve automatic reposition

Related References	
2.2.65	Procedure 2.2.65, Reactor Equipment Cooling Water System

Related Skills (K/A)	
400000.K6.05	Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: (CFR: 41.7 / 45.7) Motors (2.8/2.9)

QUESTION: 27 14039 (1 point(s))

The plant was at power with Reactor Equipment Cooling (REC) pumps A, B, and D running and REC pump C tagged out for maintenance. The B REC pump then tripped due to an electrical fault in the motor. REC pressure lowered to 40 psig before increasing and stabilizing two (2) minutes later at 53 psig.

Which of the following loads **CAN** be supplied with REC?

a. "A" Drywell Fan Coil Unit

b. "A" Station Air Compressor

c. "A" Control Rod Drive pump

d. Northwest Quad Fan Coil Unit.

ANSWER: 27 14039

d. Northwest Quad Fan Coil Unit.

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 overriden. This will isolate REC to the non-critical loops/components. The fan coil unit is the only load listed supplied from the critical loop.

Answer source: 2.2.65, p. 14, section 2.5
2.2.65, p. 15, sections 2.9 & 2.10

Distractors:

a. The drywell fancoil will remain isolated because REC pressure remains below the isolation setpoint.
b. REC flow to the air compressor will remain isolated because REC pressure remains below the isolation setpoint.
c. REC flow to the CRD pump will remain isolated because REC pressure remains below the isolation setpoint.

Source: Modified *Original Question 5279*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
28	14024	00	02/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM

Related Lessons
COR0022002 OPS REACTOR MANUAL CONTROL SYSTEM

Related Objectives	
COR0022002001150E	Given plant conditions related to RMCS and/or RPIS, determine if any of the following should occur: Control rod drift alarm
COR0022002001010I	State the purpose of the following items related to the Reactor Manual Control System and/or the Rod Position Information System: Rod Drift Alarm Test Switch

Related References	
6.CRD.303	CONTROL ROD WITHDRAWAL/OPERABILITY TEST MODE 3, 4, AND 5

Related Skills (K/A)	
201002.A4.03	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Rod drift test switch (2.8/2.8)

QUESTION: 28 14024 (1 point(s))

What represents the **MINIMUM** action(s) required to generate a rod drift alarm?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Rod drift test switch momentarily held to test position.

b. Rod drift test switch held in the test position while control rod is inserted one notch.

c. Rod drift test switch held in the test position while a control rod is inserted and then withdrawn one notch.

d. Control rod inserted one notch and the rod drift test switch is momentarily taken to test while the amber rod settle light is energized.

ANSWER: 28 14024

b. Rod drift test switch held in the test position while control rod is inserted one notch.

Any Rod movement that leaves an even reed switch or picks up an odd reed switch with the Rod Drift Alarm Test switch in test generates a Rod Drift Alarm.

<u>Answer source:</u> COR002-20-02, p. 18, section 4

Distractors:

a. Just Placing the Rod Drift Alarm Test switch to TEST does not generate a rod drift alarm.

c. While this would generate a rod drift alarm, the rod need only be either inserted OR withdrawn, so these actions do not represent the minimum required by the question.

d. This action may not generate a rod drift alarm, if the even reed switch for the control rod is made up before the Rod Drift Alarm Test switch is taken to TEST, no alarm would be generated.

<u>Source:</u> *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
29	1208	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022102, REACTOR PROTECTION SYSTEM

Related Lessons
COR0022102 REACTOR PROTECTION SYSTEM

Related Objectives	
COR0022102001100K	Describe the interrelationship between the RPS and the following: Primary Containment
COR0022102001050A	Briefly describe the following concepts as they apply to RPS: Logic arrangements

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram
2.2.22	Procedure 2.2.22, Vital Instrument Power System
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
212000.A2.09	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 conseq...: (CFR: 41.5 / 45.6) High containment/drywell pressure (4.1*/4.3*)

QUESTION: 29 1208 (1 point(s))

The plant is operating at 10% power and rated pressure during a plant startup with the following conditions:

- ➢ The "A" Reactor Protection System (RPS) MG has tripped due to a motor fault
- ➢ A loss of Drywell (DW) cooling has caused a slow but continuous rise in DW temperature and pressure
- ➢ Primary Containment parameters do not improve

What drywell pressure switch configuration will result in a full reactor scram and how is overfill of the RPV prevented?

A reactor scram will occur . . .

a. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

b. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-MO-20, DRIVE PRESSURE CONT VALVE must be closed.

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

d. only when <u>both</u> RPS system "B" DW pressure switches open. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

ANSWER: 29 1208

c. if <u>any single</u> RPS system "B" DW pressure switch opens. CRD-V-29, CHARGING WATER HEADER ROOT VALVE must be closed.

<u>Answer source:</u> COR002-21-02 p.12, section 3
2.1.5 p. 8, section 1.3

Distractors:

a. closing the drive pressure control valve will not prevent overfill.
b. Does not need to be both switches, closing the drive pressure control valve will not prevent overfill.
d. Does not need to be both switches.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
30	19096	00	05/22/2002	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022202, REACTOR RECIRCULATION

Related Lessons
SKL0124222 REACTOR RECIRCULATION SYSTEM COR0022202 REACTOR RECIRCULATION

Related Objectives	
COR0022202001130F	Given plant conditions, determine if any of the following should occur: Recirculation MG set scoop tube lock.
COR0022201001060A	Given plant and/or reactor recirculation system conditions, apply the design features and/or interlocks that provide for the following: MG Set Scoop Tube Lockout
SKL012422200A030I	Given plant conditions, predict changes in the following Reactor Recirculation System components/parameters: RR pump speed

Related References	
2.2.68	Procedure 2.2.68, Reactor Recirculation System Operations

Related Skills (K/A)	
202002.A4.01	Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) MG sets (3.3/3.1)

QUESTION: 30 19096 (1 point(s))

The plant is operating at 30% when the reactor operator is directed to raise power using Recirculation flow. As the controller output is raised, a momentary (1.5 seconds) loss of signal occurs from the "A" Recirculation Flow Controller. The operator continues to raise the controller output for several more seconds.

How will the "A" Recirculation MG Set be affected by this momentary loss and operator action?

a. The pump will automatically run back to ~ 22% speed.

b. A scoop tube lockup will prevent any further speed change.

c. After a 1.5 second pause, recirculation pump speed will rise for several seconds.

d. Speed will initially rise, then lower rapidly for 1.5 seconds, then rise again for several seconds.

ANSWER: 30 19096

b. A scoop tube lockup will prevent any further speed change.

Answer source: COR002-22-02, p. 35, section "d"

Distractors:

a. There is no runback.
b. Scoop tube lockup prevents any speed changes.
d. Scoop tube lockup prevents any speed changes.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
31	1744	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001030P	Describe RHR System design feature(s) and/or interlocks which provide for the following: Spray flow cooling
COR0022302001170C	Given plant conditions, determine actions necessary to place RHR in the following flowpaths: Drywell Spray

Related References	
2.2.69.3	Procedure 2.2.69.3, RHR Suppression Pool Cooling And Containment Spray
2.2.69	Procedure 2.2.69, Residual Heat Removal System

Related Skills (K/A)	
2.4.48	Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) (3.5/3.8)

QUESTION: 31 1744 (1 point(s))

Following a LOCA, the following conditions are present :

- Reactor pressure 700 psig (lowering slowly)
- RPV water level - 100 in (**wide range**, stable)
- Drywell press 11.0 psig (rising slowly)

What are the **MINIMUM** actions that are required in order to initiate Drywell Sprays?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. Place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

c. Place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

d. Depress Containment Spray Initiation Signal Reset pushbuttons, place Containment Cooling 2/3 Core Valve Control Permissive switches in OVERRIDE, place the Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

ANSWER: 31 1744

b. Place Containment Cooling Valve Control Permissive switches in MANUAL, then place Drywell Inbd (MO-31A/B) and Outbd (MO-26A/B) Spray Valve control switches in OPEN.

Answer source: 2.2.69, pp. 35 - 36, section 2.2.10

Distractors:

a. The permissive switch must be placed in MANUAL.
c. No need to place 2/3 core height in override.
d. No need to place 2/3 core height in override or reset logic.

2.4.48 Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12) as it applies to: 295024 High Drywell Pressure

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
32	4029	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	1	1	8	Multiple Choice	

Topic Area	Description
Systems	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001020A	State the electrical power supplies to the following: RHR pump motors
COR0022302001080A	Predict the consequences a malfunction of the following will have on the RHR system: A.C. electrical power (including RPS)

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System

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

QUESTION: 32 4029 (1 point(s))

The plant was at power with the breaker for RHR Pump 1C tagged out for maintenance when the following occur:

- ➢ A reactor coolant leak in the drywell results in a drywell pressure of 8 psig (slowly rising)
- ➢ A loss of 4160 VAC Switchgear Critical Bus 1F occurs

What RHR pumps/loops remain available and what operations be accomplished <u>from the Control Room</u>?

RHR . . .

a. Loop A with **one** pump is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

c. Loop B with **both** pumps is available for LPCI injection. Torus sprays **CANNOT** be established in either loop.

d. Loop A with **one** pump **AND** RHR Loop B with **one** pump are available for LPCI injection. Torus sprays are available from RHR loop A **ONLY**.

ANSWER: 32 4029

b. Loop B with **one** pump is available for LPCI injection. Torus sprays are available from RHR loop B **ONLY**.

Torus sprays are available from RHR loop B **ONLY**. The loss of power to 1F results in the loss of power to RHR pumps A and B. With C pump already out of service, the only remaining pump is RHR pump D, a B loop pump. Since the B loop injection valve is DC powered LPCI remains available from B loop. Power remains available to the containment cooling valves for the B loop so torus spray is available.

<u>Answer source:</u> COR002-23-02, pp. 18 & 19, Section 4
COR002-23-02, p. 67, Table 1

Distractors:

a. Neither RHR loop A pumps are available. The C pump is OOS for maintenance and with 1F bus deenergized no power is available to the A pump.
c. Only RHR pump B still has power availble so only one pump is available. Torus sprays are available from the B loop.
d. No Loop A pumps remain available and torus sprays are not available from RHR loop A.

Source: ***Modified original question 4029.***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
33	14028	00	04/30/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	COR0022302, RESIDUAL HEAT REMOVAL

Related Lessons
COR0022302 RESIDUAL HEAT REMOVAL

Related Objectives	
COR0022302001080N	Predict the consequences a malfunction of the following will have on the RHR system: Suction flow path
COR0022302001080K	Predict the consequences a malfunction of the following will have on the RHR system: Reactor water level

Related References	
2.2.69	Procedure 2.2.69, Residual Heat Removal System
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
203000.A2.02	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...: (CFR: 41.5 / 45.6) Pump trips (3.5/3.5)

QUESTION: 33 14028 (1 point(s))

"A" loop of RHR is in shutdown cooling at 150°F with "C" RHR pump running.

- ➢ RPV water level drops unexpectedly
- ➢ RPV water level continues to lower to -120 inches Wide Range

3 minutes later, what is the status of "A" and "C" LPCI pumps and what are the MIMIMUM actions are necessary to inject with **BOTH** pumps?
(NOTE: The choices are listed from MIMIMUM to MAXIMUM.)

a. "A" LPCI pump is running, "C" LPCI pump is idle. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START.

b. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START.

c. "A" LPCI pump is running, "C" LPCI pump is idle. Align pump suction paths to the Torus. Take the control switch for the "C" LPCI pump momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

ANSWER: 33 14028

d. Both pumps are idle. Align pump suction paths to the Torus. Take the control switches for the non-running pumps momentarily to STOP and then to START. Press SDC ISOL RESET VLV 25A button.

Answer source: 2.4SDC p. 13, Attachment 3

Distractors:

a. Both pumps trip and lock out on anti-pump. The SDC isolation must be reset. The suction path must be realigned.
b. The SDC isolation must be reset.
c. Both pumps trip and lock out on anti-pump.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
34	2127	01	03/28/2003	05/21/2003	Electrical	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022602, OPS ROD WORTH MINIMIZER

Related Lessons
COR0022602 OPS ROD WORTH MINIMIZER

Related Objectives	
COR0022602001010A	State the purpose of the following items related to the Rod Worth Minimizer: Rod Worth Minimizer System
COR00226020010300	State the design bases for the RWM system as described in the associated student text.
COR0022602001050M	Briefly describe the following concepts as they apply to the RWM: Minimize clad damage during control rod drop accident (CRDA)

Related References	
4.2	Procedure 4.2, Rod Worth Minimizer

Related Skills (K/A)	
2.1.28	Knowledge of the purpose and function of major system components and controls. (3.2/3.3)

QUESTION: 34 2127 (1 point(s))

Why is the Rod Worth Minimizer required below 10% power but **not** above 10%?

At higher power,

a. fewer rod movements occur which reduces the chances of an error.

b. the Rod Block Monitor prevents fuel damage in the event of a rod drop accident.

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

d. the effects of a rod drop accident are less due to increased moderator temperature causing lower rod worths.

ANSWER: 34 2127

c. the effects of a rod drop accident are less due to increased voiding causing lower rod worths.

Answer source: COR00-2-26-02, p. 9, Section "e"

Distractors:

a. voids reduce rod worths.
b. RBM mitigates rod withdrawal error.
d. Moderator temperature rise increases rod worths.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
35	14033	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022802, OPS STANDBY GAS TREATMENT

Related Lessons
COR0022802 OPS STANDBY GAS TREATMENT

Related Objectives	
COR0022802001100K	Predict the consequences of the following on the Standby Gas Treatment system: Z sump failures

Related References	
2.2.73	Procedure 2.2.73, Standby Gas Treatment System

Related Skills (K/A)	
261000.A1.01	Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: (CFR: 41.5 / 45.5) System flow (2.9/3.1)

QUESTION: 35 14033 (1 point(s))

The plant was operating at power when an LOCA occurred. The reactor building ventilation isolated and SGT automatically started. Additional plant failures occurred that resulted in the failure of Z-sump pumps and a subsequent high level in the Z-sump.

What is the potential effect on SGT?

a. SGT flow decrease

b. Inlet HEPA filter moisture damage

c. Moisture impingment on the SGT fans

d. Increased iodine carryover at the SGT train outlet

ANSWER: 35 14033

a. SGT flow decrease.

The discharge lines have drain lines that are connected to the Z sumps located at the base of the ERP. These SGT discharge lines can become blocked by excessive water level in the Z sump. If water collects in the 10" underground lines, SGT discharge flow may be restricted. Reduced SGT system flow effects the operability of the SGT systems.

Answer source: COR002-28-02, p. 23 & 24, section 3

Distractors:

b. High Z-sump level would not increase the moisture at the inlet of the SGT train.
c. A high level in the Z-sump would not impact the SGT fan.
d. A high level in the Z-sump would not result in increased iodine at the outlet of the train. The reduced flow through the train may even slightly reduce iodine at the outlet.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
36	14025	00	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0022902, STANDBY LIQUID CONTROL

Related Lessons
COR0022902 STANDBY LIQUID CONTROL

Related Objectives	
COR0022902001120A	Briefly describe the relationships that exist between the SLC system and the following: Core Spray line leak detection

Related References	
COR0022902	SLC
2.2.9	Procedure 2.2.9, Core Spray System

Related Skills (K/A)	
211000.K1.09	Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Core spray system: Plant-Specific (3.2*/3.4*)

QUESTION: 36 14025 (1 point(s))

Where do the Core Spray Line Break Detection differential pressure switches (dPIS-43A/B) connect?

The high pressure side of the pressure switch is connected to the SLC sparger to sense pressure . . .

a. below the core plate and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

b. below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

c. above the core plate in the core bypass region and the low pressure side of the switch senses pressure upstream of the Core Spray injection check valve.

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

ANSWER: 36 14025

d. above the core plate in the core bypass region and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve.

Downstream of the manual isolation valve (14A/B) each Core Spray system has an instrument line that is connected to the low pressure side of the Core Spray Line Break Detection differential pressure switch. The high pressure side of the dPIS is connected to the Standby Liquid Control "outer" pipe, which detects the pressure in the bypass region above the Core Plate.

Answer source: COR002-29-02 Figure 7

Distractors:

a. The high pressure side senses above the core plate NOT below the core plate and the low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.
b. The high pressure side senses above the core plate NOT below the core plate.
c. The low pressure side of the switch senses pressure downstream of the drywell Core Spray manual isolation valve NOT upstream of the injection check valve.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
37	5348	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM

Related Lessons
COR0023002 SOURCE RANGE MONITOR SUBSYSTEM

Related Objectives	
COR0023002001060F	Describe the SRM system design features and/or interlocks that provide for the following: IRM/SRM interlock

Related References	
4.1.1	Procedure 4.1.1, Source Range Monitoring System

Related Skills (K/A)	
215004.K3.02	Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: (CFR: 41.7 / 45.4) Reactor manual control: Plant-Specific (3.4/3.4)

QUESTION: 37 5348 (1 point(s))

A Reactor Startup is in progress with the following conditions:

- ➢ Power is rising with a stable, positive period
- ➢ SRM detectors are withdrawn except for SRM "A" which fails to withdraw
- ➢ The SRM UPSCALE OR INOPERATIVE alarm has been received
- ➢ The SRM is **NOT** bypassed

As power continues to rise, what is the **FIRST** point that rods will be able to be withdrawn?

a. Associated IRMs are on Range 3 or higher.

b. Associated IRMs are on Range 8 or higher.

c. Associated IRMs are on Range 9 or higher.

d. The Mode switch is placed to RUN.

ANSWER: 37 5348

b. Associated IRMs are on Range 8 or higher.

The SRM Upscale or Inop Rod Block is bypassed when all associated IRM's are selected to range 8 or above.

Answer source: 4.1.1 p. 5, step 1.2.2

Distractors:

a. Range 3 bypasses the detector withdrawal permissive interlock of 100 cps.
c. Range 9 has no bypass functions.
d. While RUN bypasses all SRM Interlocks/Trips, Range 8 will be achieved first.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
38	19084	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023102, TRAVERSING IN-CORE PROBE

Related Lessons
COR0023102 OPS TRAVERSING IN-CORE PROBE

Related Objectives	
COR0023102001110A	Describe the TIP system design features and/or interlocks that provide for the following: Primary containment isolation
COR0023102001130C	Given a TIP system control manipulation, predict and explain the changes in the following parameters: Valve status
COR0023102001140H	Predict the consequences of the following on the TIP system: High primary containment pressure
COR0023102001160B	Given plant conditions, determine if any of the following TIP actions should occur: Ball valve closure

Related References	
4.1.4	Procedure 4.1.4, Traversing In-Core Probe System

Related Skills (K/A)	
215001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: (CFR: 41.7 / 45.7) Primary containment isolation system: Mark-I&II (Not- BWR1) (3.1/3.4)

QUESTION: 38 19084 (1 point(s))

The plant is at 100% power with TIP traces in progress. Only "A" TIP machine is being used at this time. Currently, TIP "A" has reached the Core Bottom Limit and is moving at slow speed to the Core Top Limit. The IN-CORE light is ON.

A reactor scram due to low RPV water level occurs. One (1) minute later, an operator observes:

- TIP valve indication on Containment Isolation display (Panel 9-3) is RED
- **IN-SHIELD** light for TIP "A" is ON at Panel 9-13
- Drywell pressure is normal

What action is ***required***?

a. Fire TIP "A" shear valve.

b. Close TIP "A" ball valve.

c. Manually retract TIP "A" to fire the shear valve.

d. Manually retract TIP "A" to close the ball valve.

ANSWER: 38 19084

b. Close TIP "A" ball valve.

If red light (Panel 9-3) stays on, at least one TIP ball valve has not closed. After automatic withdrawal of the TIP on the PCIS group 2 isolation signal, the ball valve failed to automatically close. This failed automatic action requires immediate operator action to manually perform the ball valve closure. The procedure directs the operator to attempt to manually retract TIP. Since the TIP is already retracted (IN-SHIELD light is on), this action is not necessary. If ball valve cannot be closed and there are indications of a reactor coolant leak in drywell (as evidenced by the high drywell pressure) then fire appropriate shear valve by operating appropriate keylock switch.

Answer source: 4.1.4 p. 2, step 2.10
4.1.4 p. 8, step 6.3

Distractors:

a. There is no indications of a LOCA and no attempt has yet been made to close the ball valve.
c. The TIP has already retracted and retracting the TIP does not fire the shear valve.
d. The TIP has already retracted.

Source: Direct from Bank

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
39	14004	00	03/21/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL

Related Lessons
COR0023202 OPS REACTOR VESSEL LEVEL CONTROL

Related Objectives	
COR0023202001070D	Given a RVLC system control manipulation, predict and explain the changes in the following parameters: Controller Indications
COR0023202001060C	Predict the consequences of the following on the RVLC system: Control Signal Failure/Track and Hold

Related References	
2.4RXLVL	RPV WATER LEVEL CONTROL TROUBLE
2.2.28.1	Procedure 2.2.28.1, Feedwater System Operation

Related Skills (K/A)	
259002.A1.04	Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) Reactor water level control controller indications (3.6/3.6)

QUESTION: 39 14004 (1 point(s))

Given the following conditions:

- ➢ Reactor power is 90%
- ➢ Reactor water level is 35"
- ➢ RVLC is in 3 element
- ➢ The Master Controller is in BAL the tape setpoint is 35"
- ➢ The selected level instrument **INSTANTANEOUSLY** fails downscale

What is the current configuration of RFC-CS-RFPTA (RFPT A M/A station)?

The RFPT M/A station shifts to ___________ with controller output _____________ controller output prior to the event.

a. Manual Demand Mode (MDEM)
the same as

b. Manual Direct Valve Positioning Mode (MDVP)
the same as

c. Manual Demand Mode (MDEM)
higher than

d. Manual Direct Valve Positioning Mode (MDVP)
higher than

ANSWER: 39 14004

a. Manual Demand Mode (MDEM)
the same as

Answer source: New Lovejoy training material (no electronic version). New electronic versions of 2.2.28.1 and 2.4RXLVL not yet available without exam compromise.

Distractors:

b. The controller shifts to MDEM mode.
c. The output won't change.
d. The controller shifts to MDEM mode and the output won't change.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
40	14014	00	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	COR0023402, Alternate Shutdown (LO)

Related Lessons
COR0023402 Alternate Shutdown (LO)

Related Objectives	
COR00234020010700	State the design bases for the ASD system as described in the associated Student Text.
COR00234020010100	State the purpose of the Alternate Shutdown system.

Related References	
5.1ASD	Shutdown From Outside The Control Room

Related Skills (K/A)	
295016.AK3.03	Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6) Disabling control room controls. (3.5/3.7*)

QUESTION: 40 14014 (1 point(s))

When the control room is evacuated, why are the ASD panel isolation switches placed in the ISOLATE Position?

a. To prevent spurious equipment operation.

b. To ensure automatic operation of ECCS remains available.

c. To isolate circuits to meet divisional physical separation criteria.

d. To prevent overloading the associated DG during a design basis LOCA.

ANSWER: 40 14014

a. To prevent spurious equipment operation.

Answer source: COR002-34-02 p. 11, section 4

Distractors:

b. Automatic operation of ECCS does NOT remain available.
c. Operation of these switches has nothing to do with divisional separation.
d. Operation of these switches has nothing to do with diesel loading.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
41	8970	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	INT0070501, OPS Introduction to Technical Specifications

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications

Related Objectives	
INT00705010010800	From memory, state each CNS Safety Limit and discuss the basis for each of the Safety Limits.

Related References	
2.1.1.1	With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow: THERMAL POWER shall be < 25% RTP.

Related Skills (K/A)	
290002.K5.07	Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS: (CFR: 41.5 / 45.3) Safety limits (3.9/4.4)

QUESTION: 41 8970 (1 point(s))

The plant was operating at 100% power when a DEH failure results in a reactor pressure reduction. Reactor pressure decreases to 700 psig and reactor power decreases to 65%. The Group 1 isolation fails to actuate. The operating crew scrams the reactor and manually closes the Main Steam Isolation Valves.

What is a potential consequence of this event?

a. Increased likelihood of thermal hydraulic instabilities.

b. The linear heat generation rate limit for some fuel is exceeded.

c. The average planar linear heat generation rate limit is exceeded.

d. The potential is created for radioactive release in excess of 10CFR100 limits.

ANSWER: 41 8970

d. The potential is created for radioactive release in excess of 10CFR100 limits.

The scenario given represents the violation of the fuel integrity safety limit. Reactor power is greater than 25% with reactor pressure less than 785 psig. Exceeding a safety limit may cause fuel damage and create the potential for radioactive releases in excess of 10CFR100.

Answer source: Safety Limit Violation Bases p. B 2.0-5

Distractors:

a. The likelihood of thermal hydraulic instabilities is actually reduced by the decrease in core inlet subcooling casued by the rapid pressure reduction.
b. This reduction in power is global resulting in a power reduction for each core bundle. This would increase the margin to LHGR limit.
c. With the global reduction in power average linear heat generation rate would decrease and put operation farther away from the limit.

Source: ***Modified from 8970***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
42	3995	01	12/31/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	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	
INT0070502001060A	From memory, for MODES 1 and 2, state the actions required in less than one hour for: two or more control rod scram accumulators inoperable with the reactor steam dome pressure less than or equal to 940 psig (LCO 3.1.5 B.1).

Related References	
3.1.5	Control rod scram accumulators

Related Skills (K/A)	
295022.AK2.03	Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: (CFR: 41.7 / 45.8) Accumulator pressures. (3.4/3.4)

QUESTION: 42 3995 (1 point(s))

Given the following conditions:

- A Reactor startup **AND** heatup is in progress
- Reactor power is 3%
- Reactor Steam Dome Pressure is 835 psig
- Control Rod Drive Hydraulic Pump 1A trips and will not restart
- Control Rod Drive Hydraulic Pump 1B will not start

What action(s) are required by Technical Specifications for these conditions?

When the accumulator pressure is < 935 psig for . . .

a. **ANY** control rod, immediately declare the associated Control Rod inoperable.

b. **ANY** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

d. one **withdrawn** control rod, restore Charging Header pressure to greater than 940 psig within 20 minutes.

ANSWER: 42 3995

c. one **withdrawn** control rod, immediately place the Reactor Mode Switch in SHUTDOWN.

Tech Spec 3.1.5 Condition C applies. With RPV Pressure < 900 psig, withdrawn rods with inoperable accumulators may fail to scram under low pressure conditions and must be immediately inserted. Since the rod cannot be inserted without drive pressure, Condition D applies and the Reactor must be scrammed.

Answer source: Actions per LCO 3.1.5 for reactor pressure < 900 psig (RA C.1 & D.1).

Distractors:

a. This is the action for a slow control rod with an inoperable accumulator.
b. The control rod must be withdrawn to require the scram.
d. This is the action if Reactor Pressure is > 900 psig.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
43	14048	00	03/28/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	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	
INT00705040010200	Discuss the applicable Safety Analysis in the Bases associated with each Section 3.3 Specification.

Related References	
3.3.2.2	Feedwater and main turbine high water level trip instrumentation

Related Skills (K/A)	
295014.AK3.01	Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.5 / 45.6) Reactor SCRAM. (4.1*/4.1)

QUESTION: 43 14048 (1 point(s))

Why does Technical Specifications require the main turbine to trip on high reactor water level?

a. To indirectly prevent damage to the Moisture Separators by low enthalpy fluid during a failure of selected RPV water level instrument.

b. To prevent ECCS equipment damage from missiles created by main turbine failure during a feedwater controller maximum demand failure.

c. To ensure flow induced vibration of the main steam lines remains within analytical limits during a failure of selected RPV water level instrument.

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

ANSWER: 43 14048

d. To indirectly provide a reactor scram to mitigate the reduction in MCPR during a feedwater controller maximum demand failure.

Per 3.3.2.2 bases "The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event. The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR."

Answer source: Tech Spec Bases p. 3.3-55, Safety Analysis

Distractors:

a. This is not the basis.
b. The missile damage potential is not assumed by the accident analysis.
c. There is no flow induced vibration related to the main turbine trip, but is related to Recirculation flow mismatch.

Source: *New*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
44	14042	00	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

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

Related Objectives	
INT00705050010100	Given a set of plant conditions, recognize non-compliance with a Chapter 3.4 LCO.
INT00705050010300	Given a set of plant conditions that constitutes non-compliance with a Chapter 3.4 LCO, determine the ACTIONS that are required.

Related References	
6.LOG.601	Daily Surveillance Log (Tech Specs)

Related Skills (K/A)	
268000.K3.04	Knowledge of the effect that a loss or malfunction of the RADWASTE will have on following: (CFR: 41.5 / 45.3) Drain sumps (2.7/2.8)

QUESTION: 44 14042 (1 point(s))

The plant was operating at power when the Sump F Totalizer (RW-FQ-527) failed.

What is required?

Place/Leave . . .

a. **either** F-1 or F-2 sump pumps in AUTO and repair the totalizer within 4 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

b. **both** F-1 and F-2 sump pumps in AUTO and repair the totalizer within 12 hours or be in mode 3 within 12 hours and mode 4 within 36 hours.

c. **either** F-1 or F-2 sump pump switches in PULL-TO-LOCK and leave the other pump in AUTO. Record each time the pump in AUTO pumps.

d. **both** F-1 and F-2 sump pump switches to PULL-TO-LOCK. At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation.

ANSWER: 44 14042

d. F-1 and F-2 sump pump switches to PULL-TO-LOCK.

At 8 hour intervals and also when Sump F high alarm is received, pump the sump using one pump and record seconds of operation. When the Sump F Totalizer failed, 6.LOG.601 requires that the total gallons be calculated per Sump F Totalizer table. This table directs that both sump pump switches be placed in Pull-To-Lock and on 8 hour intervals and also when Sump F high alarm is received, pump sump using one pump and time seconds of operation.

Answer source: 6.LOG.601 pp. 9 (Note a) & 10 (DETERMINATION OF TOTAL GALLONS WITH FAILED SUMP F TOTALIZER)

Distractors:

a. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
b. The pumps are not placed/left in AUTO and 30 days are allowed to repair the totalizer.
c. Neither pump is left in AUTO.

<u>Source:</u> ***New***

<u>Provide to Candidate:</u> T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
45	6226	03	05/16/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320128, CNS Abnormal Procedures (RO) Containment

Related Lessons	
INT0320128	CNS Abnormal Procedures (RO) Containment

Related Objectives	
INT0320128J0J0100	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	
2.3_9-3-2	PANEL 9-3 - ANNUNCIATOR 9-3-2
2.4PC	PRIMARY CONTAINMENT CONTROL

Related Skills (K/A)	
295010.AK1.03	Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.8 to 41.10) Temperature increases. (3.2/3.4)

QUESTION: 45 6226 (1 point(s))

The plant is operating at 100% power when the following conditions exist:

- Annunciator H-1/A-2 DRYWELL ZONE 1 HIGH TEMP is alarming
- Annunciator H-1/A-1, DRYWELL FCU A HIGH DISCH TEMP is alarming
- Annunciator 9-5-2/F-3, DRYWELL HIGH PRESSURE is alarming
- Two (2) of the Drywell FCU's have tripped

What actions are required of the operator as identified in Abnormal Procedure 2.4PC, PRIMARY CONTAINMENT CONTROL?

a. Maintain drywell pressure ≤ 0.75 psig by venting via SGT, **ONLY**.

b. Start tripped drywell FCUs by placing their control switches to OVERRIDE, **ONLY**.

c. Maintain drywell pressure ≤ 0.75 psig by venting via SGT **AND** if drywell pressure cannot be maintained below 1.5 psig, scram the Reactor.

d. Start tripped drywell FCUs by placing their control switches to OVERRIDE **AND** maintain drywell pressure ≤ 0.75 psig by venting via SGT.

ANSWER: 45 6226

c. Maintain drywell pressure ≤ 0.75 psig by venting via SGT **AND** if drywell pressure cannot be maintained below 1.5 psig, scram the Reactor.

Answer source: PR 2.4PC

Distractors:

a. The ONLY does not include the required scram.
b. Cannot place FCUs in OVERRIDE until in EOP-3A and does not include required scram.
d. Incorrect as the does not include the required scram and cannot place FCUs in OVERRIDE until in EOP-3A.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
46	5247	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.

Related References	
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 46 5247 (1 point(s))

The plant is at 35% power when Turbine vibration required the operator to insert a manual scram and trip the turbine. The following conditions exist one (1) minute after the operator has depressed both manual reactor scram pushbuttons:

- Turbine Bypass Valves are 75% open **(stable)**
- Main Steam Isolation Valves (MSIVs) are open
- RPV Narrow Range level is 25" **AND steady**
- RPV pressure is 940 psig **AND steady**

Which procedure(s) must be executed?

2.1.5 "Reactor Scram" . . .

a. **ONLY**.

b. **AND** EOP-1A "RPV Control" **ONLY**.

c. **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)" **ONLY**.

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

ANSWER: 46 5247

d. **AND** EOP-1A "RPV Control" **AND** EOP-6A "Reactor Pressure/Power (Failure to Scram)" **AND** EOP-7A "Reactor Level (Failure to Scram)".

Bypass valves at 75% open is approximately 19% power. 19% power after a scram is an entry condition to 1A. EOP 1A directs 6A and 7A to be entered.

Answer source: EOP-1A, 2.1.5 p. 1

Distractors:

a. EOP 1A must be entered.
b. Entry into 6A & 7A is required.
c. Entry into 1A is required.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
47	13407	0	04/12/2001	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050010900	Identify any EOP support procedures addressed in Flowchart 1A and apply any associated special operating instructions or cautions.

Related References	
5.8.2	Procedure 5.8.2, Alternate Emergency Depressurization Systems (Table 2)

Related Skills (K/A)	
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10) (2.9/3.3)

QUESTION: 47 13407 (1 point(s))

While responding to a LOCA, the RCIC overspeed trip must be reset to allow it to be used as an injection system.

- ➢ The TSC is NOT operational.
- ➢ Several Reactor Building ARMs that an operator must pass within 10 feet of to get to the area are alarming and indicate upscale.
- ➢ Both high range Drywell radiation monitors read 1.5E4 R/hr.

In addition to standard RP practices, what additional (if any) MINIMUM requirement must be met to dispatch an operator to perform this task?
(NOTE: The choices are arranged in MIMIMUM to MAXIMUM order.)

a. No additional requirements must be met.

b. A survey instrument that monitors radiation dose rates must be taken.

c. A RP Technician must accompany the operator.

d. The TSC must be declared operational.

ANSWER: 47 13407

d. The TSC must be declared operational.

If DRYWELL RAD MONITOR RMA-RM-40A or DRYWELL RAD MONITOR RMA-RM-40B (Panel 9-02) is reading ≥ 1E4 rem/hour, entry into Secondary Containment is prohibited until TSC is operational and personnel can be dispatched per Procedure 5.7.15. The operator cannot be dispatched until the TSC is operational because both drywell radiation monitors are above 1E4 rem/hr.

Answer source: 5.8.2 p. 12, section 5.1

Distractors:

a. The TSC must also be operational to dispatch the operator.
b. The TSC must also be operational to dispatch the operator.
c. The TSC must also be operational to dispatch the operator.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
48	14000	00	02/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
COR0020302	CONTAINMENT
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
COR0020302001210B	Given plant conditions, determine if the following should have occurred: Any of the PCIS group isolations.
INT0080605001010A	List the entry conditions of Flowchart 1A: Describe the importance of each in an emergency situation.

Related References	
2.1.22	Procedure 2.1.22, Recovering From A Group Isolation

Related Skills (K/A)	
2.4.2	Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) (3.9/4.1)

QUESTION: 48 14000 (1 point(s))

RPV water level lowers to 20 inches below the value that is an entry condition for EOP-1A, "RPV Control". No other EOP entry conditions are satisfied.

What group isolations have automatically occurred?

Group 2 . . .

a. **only**.

b. and Group 3 **only**.

c. and Group 3 and Group 6 **only**.

d. and Group 1 and Group 3 and Group 6.

ANSWER: 48 14000

c. and Group 3 and Group 6 **only**.

Answer source: 2.1.22 p. 7, section 5.3
2.1.22 p. 10, section 6.3
2.1.22 p. 17, section 9.6

Distractors:

a. Group 3 and Group 6 also occur.
b. group 6 also occurs.
d. Group 1 does not occur.

2.4.2 Knowledge of system set points / interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8) as it applies to: 295009 Low Reactor Water Level

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
49	2403	01	05/19/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	COR0020902, Digital Electro-Hydraulic Control

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001040C	Describe how the DEH control system operates to control the following: Reactor/turbine pressure regulating system load set/reference
COR0020902001040G	Describe how the DEH control system operates to control the following: Reactor steam flow
COR0020902001040H	Describe how the DEH control system operates to control the following: Main Turbine steam flow

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System

Related Skills (K/A)	
295025.EA1.02	Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6) Reactor/turbine pressure regulating system (3.8/3.8)

QUESTION: 49 2403 (1 point(s))

With DEH load limit set at 300 MW, reactor power is raised from 250 to 400 MWE.

What will be the final Steam Flow and Main Turbine Bypass Valve (BPV) conditions?

a. Rx. steam flow = turbine steam flow, BPVs closed.

b. Rx. steam flow < turbine steam flow, BPVs closed.

c. Rx. steam flow > turbine steam flow, BPVs partially open.

d. Rx. steam flow = turbine steam flow, BPVs partially open.

ANSWER: 49 2403

c. Rx. steam flow > turbine steam flow, BPVs partially open.

As power increases, reactor pressure will rise. The rise in reactor pressure will raise equalizing header pressure, causing governor valves to open until load limit is met. Then, the increase in pressure will cause the Main Turbine Bypass Valves to open to maintain pressure.

Answer source: COR002-09-02, p. 9, section "c."

Distractors:

a. Turbine steam flow is limited to 300 MW, BPVs will be open.
b. Rx steam flow will be > turbine steam flow, BPVs will be open.
d. Turbine steam flow is limited to 300 MW.

Source*: Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
50	19034	01	03/21/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Lessons	
INT0080618	EOP AND SAG GRAPHS
INT0080605	OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE

Related Objectives	
INT00806050011000	Given plant conditions and EOP flowchart 1A, RPV CONTROL, determine required actions.
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.
INT00806050011200	Given plant conditions, assess if RPV water level can be determined or not.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 1A, RPV CON	RPV Control

Related Skills (K/A)	
226001.A1.02	Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: (CFR: 41.5 / 45.5) Containment/drywell temperature (3.4/3.5)

QUESTION: 50 19034 (1 point(s))

A Loss of Coolant Accident has occurred with the following conditions:

➢ Reactor pressure	270 psig (lowering at 5 psig/minute)
➢ Indicated water level	-120" (Wide range, steady)
➢ Drywell pressure	5.5 psig (rising slowly)
➢ Drywell temperature	350° F (all points) (steady)

What is the status of Wide Range Reactor Level Instrumentation ***now*** and what effect (if any) will drywell sprays have on ***future*** availability?

Wide Range Reactor Level Instrumentation . . .

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

b. can be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

c. **CANNOT** be used for trending.
Initiation of drywell sprays will restore instrument availability.

d. **CANNOT** be used for trending.
Initiation of drywell sprays will have **NO** effect on future instrument availability.

ANSWER: 50 19034

a. can be used for trending.
Initiation of drywell sprays will help maintain instrument availability.

Indicated WR Level is above the minimum Indicated Level of EOP Graph 15 for 350° F so the instrument can be used for trending purposes. Initiation of DW sprays cools containment and prevents intrusion into the unsafe region of reactor saturation graph (EOP Graph 1), in addition DW sprays and their cooling effect on DW temperature ensure that the instrumet **stays** in the safe region of Graph 15.

Answer source: EOP Graph 1 & EOP Graph 15, effects of drywell spray

Distractors:

b. Drywell sprays will maintain Instrument availability.
c. The instruments are currently available for trending.
d. The instruments are currently available for trending and drywell sprays will maintain Instrument availability.

<u>Source:</u> ***Direct from bank***

<u>Provide to Candidate:</u> EOP Graphs

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
51	14001	00	02/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010600	List the methods of alternate rod insertion.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295015.AK2.04	Knowledge of the interrelations between INCOMPLETE SCRAM and the following: (CFR: 41.7 / 45.8) RPS (4.0/4.1)

QUESTION: 51 14001 (1 point(s))

The plant was operating at 100% power with a half scram on RPS A due to a relay failure and 1B CRD pump tagged out of service. Subsequently, power was lost to RPSPP 1B. The following conditions now exist:

- Many control rods failed to insert
- Reactor power is 5%
- 4160 VAC Bus 1F is de-energized

What method can the crew use to successfully insert the control rods that failed to insert on the scram?

a. Individually scram control rods.

b. Drain the SDV and manually scram.

c. Manually vent CRDM over-piston area.

d. Insert control rods using emergency override switch.

ANSWER: 51 14001

c. Manually vent CRDM over-piston area.

The multiple RPS failures result in the inability to reset the scram. In addition, having the 1B CRD pump out of service in conjunction with the loss of power to the operating CRD pump precludes the use of RMCS to drive control rods. Venting the over piston area of the CRDM is the only action listed that is consistent with 5.8.3, Alternate Rod Insertion Methods.

Answer source: 5.8.3 flowchart, path "B"

Distractors:

a. The scram valves cannot be closed due to the RPS failures.
b. Draining the SDV and manually scramming would require that the scram valves be closed which is not possible with the RPS failures.
d. RMCS cannot be used to drive control rods because there are no operating CRD pumps.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
52	16472	01	03/24/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Lessons	
INT0080606	FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCRAM

Related Objectives	
INT00806060010600	List the methods of alternate rod insertion.
INT00806060011000	Given plant conditions and ESP 5.8.3, ALTERNATE ROD INSERTION METHODS, determine which methods would successfully insert control rods.
INT00806060010900	Identify any EOP support procedures addressed in Flowchart 6A and apply any associated special operating instructions or cautions.

Related References	
5.8.3	Procedure 5.8.3, Alternate Rod Insertion Methods

Related Skills (K/A)	
295037.EK2.05	Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: (CFR: 41.7 / 45.8) CRD hydraulic system (4.0/4.1)

QUESTION: 52   16472 (1 point(s))

The plant scrammed from full power, but many control rods failed to insert (see following figure). Control rods will be inserted using the Reactor Manual Control System.

Which one of the following rod insertion sequences should be used?
(NOTE: Blank positions on the figure indicate control rod is fully inserted.)

Fully insert rod . . .

a. 26-27, then fully insert 26-39, then 30-27, then 34-27 and continue to insert rods that have high worth.

b. 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

c. 30-27, then fully insert 18-27, then 30-35 and continue to criss-cross the core outward from the center of the core.

d. 30-27, then fully insert 22-27, then 22-23 and continue this spiral pattern inserting the most withdrawn control rods first.

	02	06	10	14	18	22	26	30	34	38	42	46	50
51													
47													
43													
39				10			32	36					
35					12	26		36					
31				12		36			10				
27					12	48	46	42	32				
23						48			24				
19					10		34	12					
15													
11													
07													
03													

ANSWER: 52 16472

b. Fully insert rod 26-27, then fully insert 22-23, then 22-31 and continue this spiral pattern outward from the center of the core.

Under ATWS conditions, control rod insertions should begin in the center of the core, and proceed to every other control rod in an outward spiral pattern.

Answer source: 5.8.3 p. 4, section 4.2

Distractors:

a. Skips control rods, this method has no real pattern to it.
c. Does NOT start with the center rod and does not insert every other rod by geometric location.
d. Does NOT start with the center rod and does not insert every other rod by geometric location.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
53	16485	02	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	6	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM

Related Lessons	
INT0080610	OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM
INT0080612	FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM
INT0080602	OPS FLOWCHART ORGANIZATION AND STRUCTURE

Related Objectives	
INT00806100010900	Given an EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM) step, state the reason for the actions contained in the step.
INT00806120010200	Describe the conditions required to assure adequate core cooling while flooding the core during a failure to scram transient.
INT00806020010800	List the three mechanisms used in the EOP flowcharts to assure adequate core cooling.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295015.AK1.04	Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM: (CFR: 41.8 to 41.10) Reactor pressure: Plant-Specific. (3.8/3.8)

QUESTION: 53 16485 (1 point(s))

Which one of the following conditions, by itself, assures fuel clad temperature does not exceed 1500°F ***<u>during an ATWS</u>***?
(Note: All RPV levels are as INDICATED on the Fuel Zone instruments.)

Reactor Pressure . . .

a. 197 psig, ***indicated*** RPV level -50 inches, Two (2) SRVs open.

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

c. 452 psig, ***indicated*** RPV level -66 inches, One (1) SRV open.

d. 790 psig, ***indicated*** RPV level -60 inches, **NO** SRVs open.

ANSWER: 53 16485

b. 351 psig, ***indicated*** RPV level -55 inches, Three (3) SRVs open.

Actual level in all 4 choices is below -25 inches for adequate core cooling with level/steam cooling. Minimum Steam Cooling Pressure met only in choice "b" (MSCP for 3 SRVs is 280 psig).

<u>Answer source:</u> EOP flowchart 7A steps FS/L-12 & FS/L-17
EOP Flowcharts 7A & 7B, Table 14
INT008-06-06 p.20

Distractors:

a. Below -25" actual, below MSCP.
b. Below -25" actual, below MSCP.
d. Below -25" actual, below MSCP.

<u>Source:</u> *Modified*

<u>Provide to Candidate:</u> EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
54	54	0	05/19/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM

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

Related Objectives	
INT00806100010700	Identify any EOP support procedure addressed in Flowchart 7A and apply any associated special operating instructions or cautions.
INT00806100010800	Given plant conditions and EOP flowchart 7A, RPV LEVEL (FAILURE TO SCRAM), determine required actions.

Related References
NONE

Related Skills (K/A)	
295031.EA1.12	Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) Feedwater. (3.9/4.1*)

QUESTION: 54 54 (1 point(s))

A Manual reactor scram was inserted from rated power. The following conditions exist:

- ➢ Reactor power is 30%
- ➢ Reactor water level is 35"
- ➢ The Master Controller is in BAL the tape setpoint is 35"

What is the correct sequence of actions to Stop and Prevent injection from the Reactor Feedwater pumps?

a. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Demand Mode (MDEM) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

b. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Direct Valve Positioning Mode (MDVP) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

c. Close the Reactor Feedwater pump discharge valves and trip the Reactor Feedwater pumps. Then ensure RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) are in Manual Demand Mode (MDEM) with output at minimum.

d. Close the Reactor Feedwater pump discharge valves and trip the Reactor Feedwater pumps. Then ensure RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) are in Manual Direct Valve Positioning Mode (MDVP) with output at minimum.

ANSWER: 54 54

b. Place RFC-CS-RFPTA and RFC-CS-RFPTB (RFPT M/A stations) in Manual Direct Valve Positioning Mode (MDVP) and reduce output to minimum. Then ensure the Reactor Feedwater pump discharge valves are closed and trip the Reactor Feedwater pumps.

The feedwater pump is placed in MDVP mode and reduced to minimum before closing the discharge valves and tripping the reactor feedwater pumps. This sequence of events rapidly reduces feedwater flow.

<u>Answer source:</u> Procedure 5.8 page 4.

Distractors:

a. The controller is placed in MDVP mode.
c. The RFP is tripped after the controller is adjusted. The controller is placed in MDVP mode.
d. The RFP is tripped after the controller is adjusted. The controller is placed in MDVP mode.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
55	14047	00	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

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

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295013.AA2.01	Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Suppression pool temperature (3.8/4.0)

QUESTION: 55 14047 (1 point(s))

Given the following conditions:

- ➢ Primary Containment water level 14 feet
- ➢ Reactor pressure 800 to 1000 psig (with SRVs)

Primary Containment water level	14 feet
Reactor pressure	800 to 1000 psig (with SRVs)

Which of the following is the **LOWEST** average suppression pool water temperature which requires Emergency Depressurization?

a. 195°F

b. 205°F

c. 211°F

d. 215°F

ANSWER: 55 14047

b. 205°F

Answer source: EOP graph 7
EOP flowchart 3A step SP/T-5

Distractors:

a. Too low.
c. Too high.
d. Too high.

Source: ***New***

Provide to Candidate: EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
56	5268	02	02/26/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295012.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Drywell pressure (3.9/4.1)

QUESTION: 56 5268 (1 point(s))

Given the following conditions:

- ➢ A small coolant leak has occurred in the Drywell
- ➢ Drywell temperature is 185°F and rising slowly
- ➢ Drywell pressure is 4.0 psig and rising slowly

What action is required to control Drywell conditions?

a. Initiate drywell sprays.

b. Operate all available drywell cooling.

c. Vent primary containment with torus vent line.

d. Vent primary containment with drywell vent line.

ANSWER: 56 5268

b. Operate all available Drywell Cooling.

This action is specified in the drywell temperature leg of EOP-3A when drywell temperature cannot be maintained below 150°F.

Answer source: EOP flowchart 3A step DW/T-3

Distractors:

a. Drywell sprays are not permitted with torus pressure at the current 2.6 psig.
c. Venting the torus is not allowed with the LOCA signal present and pressure well below PCPL-A.
d. Venting the drywell is not allowed with the LOCA signal present and pressure well below PCPL-A.

Source: ***Modified 5268***

Provide to Candidate: EOP-3A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
57	5332	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS 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.
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.1.20	Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) (4.3/4.2)

QUESTION: 57 5332 (1 point(s))

Given the following conditions:

- A Loss of Coolant Accident has occurred
- Reactor pressure is 590 psig
- Drywell pressure is 6.5 psig **AND** rising
- Drywell temperature is 200° F
- Drywell Spray is in service with suction ***from the torus***
- Torus Temperature is 165° F
- Torus level is 16.8 feet **AND** rising
- Torus spray is in service

What containment spray action is required **AND** what is the bases for the action?

a. Terminate Drywell Spray to stop the water addition to the Torus.

b. Terminate Torus Spray since the Torus Spray Header is submerged.

c. Terminate Torus Spray to raise Torus pressure to drive non-condensable gases into the Drywell.

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

ANSWER: 57 5332

d. Terminate Drywell Spray because the primary containment vacuum relief system capacity has been exceeded.

Answer source: EOP flowchart 3A steps DS-1 & DS-2
INT008-06-13 p. 11

Distractors:

a. Drywell Spray water can be taken from the Torus.
b. The Spray header is submerged at 26.75'.
c. The Vacuum Breakers will not pass sufficient flow when covered.

2.1.20 Ability to execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) as it applies to: 295029 High Suppression Pool Water level / 5

Source: ***Direct from bank***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
58	5333	00	11/16/1999	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT0080613001040A	State the basis for primary containment control actions as they apply to the following: Specific setpoints
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
500000.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6) Emergency depressurization (3.1/3.9)

QUESTION: 58 5333 (1 point(s))

What hydrogen **AND** oxygen concentration values require Emergency Depressurization **AND** what is the bases for the Emergency Depressurization?

Containment hydrogen AND oxygen concentrations require an Emergency Depressurization at _________________ in order to ensure the . . .

a. 5% H_2 **AND** 6% O_2
source of hydrogen production is removed.

b. 6% H_2 **AND** 5% O_2
source of hydrogen production is removed.

c. 5% H_2 **AND** 6% O_2
Reactor is at the lowest energy state possible.

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

ANSWER: 58 5333

d. 6% H_2 **AND** 5% O_2
Reactor is at the lowest energy state possible.

Answer source: EOP flowchart 3A Table 7
INT008-06-13 p. 18

Distractors:

a,b. Combustible limits are 6% hydrogen and 5% oxygen. Depressurization does not remove the hydrogen source.
c. Concentrations are reversed.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
59	14023	00	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Emergency Operating Procedures	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011200	Given plant conditions and EOP flowchart 3A, PRIMARY CONTAINMENT CONTROL, state the reasons for the actions contained in the steps.

Related References	
PSTG	Plant Specific Technical Guideline
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

Related Skills (K/A)	
295028.EK3.05	Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.5 / 45.6) Reactor SCRAM (3.6/3.7)

QUESTION: 59 14023 (1 point(s))

When performing actions in EOP-3A "Primary Containment Control" for high drywell temperature, what is the reason for entering **AND** performing EOP-1A "RPV Control" concurrently without a specific EOP-1A entry condition being met?

Entering EOP-1A "RPV Control" and inserting a manual scram ensures . . .

a. the power produced by the reactor will be within the Primary Containment vent capability.

b. the RPV is at the lowest possible energy state before implementing more severe strategies.

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

d. the main source of potential energy addition to the Primary Containment is removed before conditions warrant Emergency Depressurization.

ANSWER: 59 14023

c. the reactor is scrammed and shutdown by control rod insertion before RPV depressurization is initiated.

Answer source: INT008-06-13 p. 22

Distractors:

a. RCIC exhaust flowrate, factors for basis of PCPL-A.
b. The "lowest energy state" is an Emergency Depressurization basis. There is no more severe strategy than Emergency Depressurization.
d. It is not a "potential" energy addition and scramming the reactor does not remove the source of the energy addition.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
60	19082	00	05/16/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130011000	Identify any EOP support procedures referenced in Flowchart 3A and apply any associated special operating instructions or cautions.
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure
5.9H2O2	Severe Accident Procedure (Primary Containment Combustible Gas Control [SAG 3])

Related Skills (K/A)	
223001.K6.04	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: (CFR: 41.7 / 45.7) Combustible gas mixing: Plant-Specific (2.8/2.8)

QUESTION: 60 19082 (1 point(s))

A LOCA has occurred and the following conditions exist:

➢ Drywell H_2	9% (rising slowly)
➢ Drywell O_2	4% (steady)
➢ Suppression Chamber H_2	4% (rising slowly)
➢ Suppression Chamber O_2	6%. (steady)
➢ Drywell Spray	Unavailable

What effect does the lack of Drywell sprays have on the Primary Containment and what action is required?

The inability to initiate drywell sprays . . .

a. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (within ODAM release rate limits) is required.

b. decreases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (within ODAM release rate limits) is required.

c. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

d. decreases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

ANSWER: 60 19082

c. increases the chance of containment damage from hydrogen deflagration. Emergency venting of the Primary Containment (exceeding offsite radioactivity release rate limits if necessary) is required.

The lack of drywell sprays prevents mixing of combustible gasses and increases gas concentrations. The lack of sprays into the atmosphere also increases flamability. These indications are sufficient for Combustiblity. SAP 5.9H2O2 requires emergency venting the PC irrespective of off-site release rates.

<u>Answer source:</u> EOP-3A; PC/G-2 and TABLE 7.
SAP5.9H202

Distractors:

a. Required to vent irrespective of release rate limit.
b. Increases chance of deflagration, required to vent irrespective of release rate limit.
d. Increases chance of deflagration.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed. 5.9H2O2

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
61	5730	03	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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

Related Objectives	
INT00806170010400	State the basis for the limits of the maximum safe operating values (MSO) as they apply to personnel protection and equipment operability.

Related References	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 61 5730 (1 point(s))

What is the basis for the value selected as the EOP Flowchart 5A Maximum Safe Operating (MSO) Radiation level?

It is based on personnel exposures exceeding the . . .

a. CNS TEDE dose limit of 3 Rem ***per quarter*** for a **one hour** stay time.

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

c. 10CFR20 planned special exposure limit of 3 rem for a **one hour** stay time.

d. 10CFR20 planned special exposure limit of 5 rem for a **two hour** stay time.

ANSWER: 61 5730

b. CNS TEDE dose limit of 4 Rem ***per year*** for a **two hour** stay time.

Answer source: INT008-06-17 pp. 10 & 11

The MSO radiation level of 1000 mr/hr (Table 10) is based on personnel exposure assuming:

- The CNS TEDE dose limit of 4 Rem (4000 mrem) per year (Procedure 9.ALARA.1) derated by 0.67 (1.5 safety factor).

- A two hour stay time. (Two hours is the maximum time an individual would be expected to remain in secondary containment during emergency conditions.)

- An assumed internal dose contribution of 20% of the external dose contribution (threshold for respiratory protection from an ALARA standpoint, Procedure 9.ALARA.5).

Or, 4000 mrem/(1.2*2*1.5 hours) = 1111 mrem /hour. A Maximum Safe Opeating radiation level of 1000 mrem/hour is specified because it is reasonably close to the 1111 mrem/hour value and much easier to remember and read. Health Physics access control determines the method of monitoring for this limit.

Distractors:

a. The quarterly limit no longer exists, but it was previously the basis for MSO radiation level, and herefore remains a plausible distractor.

c, d. MSO Radiation levels are not based on the 10CFR20 limits for planned special exposures.

2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements as it applies to 290001 Secondary Containment

Source: ***Modified from 5730***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
62	14019	00	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295038.EK3.04	Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6) Emergency depressurization (3.6/3.9)

QUESTION: 62 14019 (1 point(s))

What is the basis for performing an Emergency Depressurization during the execution of EOP 5A, **RADIOACTIVITY RELEASE CONTROL**?

Performing an Emergency Depressurization ensures the . . .

a. availability of equipment in the turbine building that may be necessary to mitigate the event is not challenged.

b. energy level of the radiation and the atomospheric dispersion factors fall within the bounds of the accident analysis.

c. isotopic mixture of radioactive materials deposited off-site will be within the bounds of the accident analysis.

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

ANSWER: 62 14019

d. lowest possible driving head and flow of primary systems that are unisolated and discharging outside of containment.

Answer source: INT008-06-17 p. 14

Distractors:

a. availability of turbine building equipment is not an EOP consideration, but sounds like the reactor building ED reason.
b. This is a reason to use the DOSE program for the projections vice the ODAM calculations, but is not the reason for the ED.
c. This is not the basis for the ED.

Source: *New*

Provide to Candidate: EOP 5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
63	14021	00	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
295036.EK3.02	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.5 / 45.6) Reactor SCRAM. (2.8/2.8)

QUESTION: 63 14021 (1 point(s))

Why is a reactor scram required before exceeding a Maximum Safe Operating Water Level if a primary system is discharging into the Reactor building (EOP-5A step SC-12)?

Scramming the Reactor . . .

a. provides mitigating action such that the condition does not pose an immediate threat to the health and safety of the public.

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

c. promptly reduces to decay heat levels the energy discharged into primary containment and reduces the driving head and flow of primary systems that are unisolated.

d. promptly places the primary system in its lowest possible energy state and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.

ANSWER: 63 14021

b. promptly reduces energy to decay heat levels and reduces the likelihood of requiring rapid depressurization of the RPV.

Answer source: INT008-06-17 p. 10, section 10

Distractors:

a. Not a bases for scramming.
c. Primary Containment Control bases.
d. Bases for emergency depressurization.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
64	16483	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

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

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

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

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 64   16483 (1 point(s))

The plant is at 90% when a RCIC steam line break occurs. Initial event conditions were:

- ➢ RCIC cannot be isolated
- ➢ RCIC temperatures in the NE Quad are 214°F and rising
- ➢ Reactor Building ventilation exhaust radiation levels are 25 mr/hr
- ➢ RCIC Room radiation levels are 300 mr/hr and rising
- ➢ A reactor scram is inserted and all control rods fully insert

Five (5) minutes later, annunciator 9-4-1/E-4 RX BLDG VENT HI HI RAD clears.

What is the required EOP response to this annunciator clearing and what is the bases for the response?

a. Restart Reactor Bldg. HVAC to ensure all radioactive discharges are elevated.

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

c. Do **NOT** restart Reactor Bldg. HVAC because EOP 1A requires a group 6 isolation.

d. Do **NOT** restart Reactor Bldg. HVAC until RP ensures normal radiation levels to minimize the spread of contamination.

ANSWER: 64 16483

b. Restart Reactor Bldg. HVAC to help return secondary containment parameters to normal.

Answer source: INT008-06-17 p. 7, section "b"

Distractors:

a. This is not a bases for restarting Reactor Bldg. HVAC
c. The Isolation would only occur on low level (3) and if so, it should be bypassed.
d. Per 2.1.22 if a Group 6 Isol had occurred, it should not be reset until Chem. and HP have ensured normal rad levels, but EOPs take precedence.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
65	19068	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

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

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	
EOP/SAG PSTG	Plant Specific Technical Guideline

Related Skills (K/A)	
2.3.11	Ability to control radiation releases. (CFR: 45.9 / 45.10) (2.7/3.2)

QUESTION: 65 19068 (1 point(s))

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: 65 19068

b. preserves personnel accessibility **AND** assures that radioactivity releases pass through a monitored release point.

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.

Answer source: INT008-06-17 p. 13, 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.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
66	14046	00	03/28/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080618, EOP AND SAG GRAPHS

Related Lessons	
INT0080618	EOP AND SAG GRAPHS

Related Objectives	
INT00806180010100	Using the graphs provided in the EOP and SAG Graphs Flowchart, determine how the shape of each curve or family of curves was determined.

Related References	
PSTG	Plant Specific Technical Guideline
INT0080618	EOP and SAG Graphs and Cautions

Related Skills (K/A)	
295030.EK1.03	Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10) Heat capacity. (3.8/4.1*)

QUESTION: 66 14046 (1 point(s))

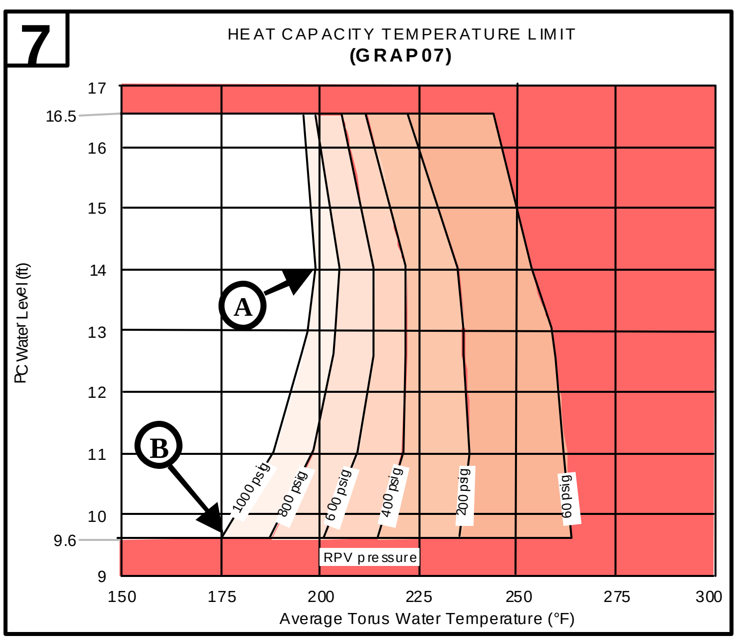
What is the bases for the shape of the Heat Capacity Temperature Limit (GRAPH 07) from point "A" to point "B"?

a. As torus level lowers, there is less backpressure on the SRV tailpipes. With lower SRV backpressure, a lower initial temperature will exceed analyzed blowdown rates during a LOCA.

b. As torus level lowers, there is less backpressure on the downcomers. With lower downcomer backpressure, a lower initial temperature will exceed peak design containment pressure during a LOCA blowdown.

c. Torus airspace volume increase has a greater effect on the limit than torus heat sink mass decrease. As torus airspace volume rises, a lower temperature is needed to heat and pressurize the airspace to PCPL-A.

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.



ANSWER: 66 14046

d. Torus heat sink mass decrease has a greater effect on the limit than torus airspace volume increase. As torus level decreases, there is less mass available to absorb the blowdown energy from the RPV, requiring a more restrictive limit.

Answer source: INT008-06-18 pp. 12 & 13

Distractors:

a. HCTL is based on not exceeding PCPL-A and has nothing to do with SRV backpressure.
b. HCTL is based on not exceeding PCPL-A and has nothing to do with downcomer backpressure.
d. Torus airspace volume increase as a lesser effect and acts to make a less restrictive limit rather than a more resrictive limit.

Source: ***New***

Provide to Candidate: EOP Graphs.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
67	4243	01	01/26/2000	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)

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

Related Objectives	
INT032010100G010M	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Surveillance Test Frequency

Related References	
0.26	Surveillance Program

Related Skills (K/A)	
2.2.12	Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) (3.0/3.4)

QUESTION: 67 4243 (1 point(s))

A monthly surveillance test was completed at 1300 on May 4th (SR 3.6.4.3.1, Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.).

In accordance with procedure 0.26 (Surveillance Program), what is the **LATEST** this test can be scheduled to be completed, without exceeding an LCO?

a. 1300 on June 4th

b. 1300 on June 5th

c. 1300 on June 11th

d. 0700 on June 12th

ANSWER: 67 4243

d. 0700 on June 12th

A deviation of 25% of the monthly surveillance interval (38.75) is allowed

Answer source: 0.26 Surveillance Program Page 10 Section 8.1.5

Distractors:

a. Using 30 day as montly with no extension.
b. Using 31 day as montly with no extension.
c. Using 38 vice 38.75.

2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13) as it applies to: 261000 SGTS

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
68	12215	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320102, CNS Administrative Procedures Site Services Procedures (Form

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure
INT032010200C010B	Discuss the following as described in Administrative Procedure 1.15, Visitor/Tour Station Access: Vital Area access criteria

Related References	
1.15	Visitor/Tour Station Access

Related Skills (K/A)	
2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10) (2.6/3.0)

QUESTION: 68 12215 (1 point(s))

Four vendor representatives were provided an escorted tour to perform a walkdown of the RHR system to assist in correcting an emergent problem. TLDs were not issued to the vendors. Following the walkdown, the following dose was received by each of the vendors on their DRDs:

- ➢ Rick Cox 106 mrem
- ➢ Ira Fox 98 mrem
- ➢ Marvin Estes 51 mrem
- ➢ Robert Smith 101 mrem

Which of the following are required?

An Exposure History Worksheet must be completed for . . .

a. **all** the visitors prior to departure.

b. **only** Rick Cox before departure.

c. **only** Rick Cox and Robert Smith before departure.

d. **only** Rick Cox, Robert Smith and Ira Fox before departure.

ANSWER: 68 12215

c. **only** Rick Cox and Robert Smith before departure.

If the dose received is > 100 mrem (1.0 mSv) by DRD, or equivalent, and no TLD was issued, Radiological Protection shall be contacted and a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing the visitor to depart. Both these individuals received greater than 100 mrem and since TLDs were not issued, a CNS-RP-10, Exposure History Worksheet, shall be completed before allowing they depart.

Answer source: 1.15 page 4, step 4.6

Distractors:

a. Ira Fox and Marvin Estes received less than 100 mrem and are not required to complete CNS-RP-10, Exposure History Worksheet.
b. Robert Smith is also required to complete CNS-RP-10, Exposure History Worksheet prior to departure.

d. Ira Fox received less than 100 mrem and is not required to complete CNS-RP-10, Exposure History Worksheet.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
69	12209	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	INT0320103, CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Lessons	
INT0320103	CNS Administrative Procedures Conduct of Operations and General Alarm Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010300E010B	Discuss the following as described in Alarm Procedure 2.3.1, General Alarm Procedure: Alarm acknowledgement

Related References	
2.3.1	Procedure 2.3.1, General Alarm Procedure
OI-07	Operations Management Expectations

Related Skills (K/A)	
2.4.31	Knowledge of annunciators alarms and indications / and use of the response instructions. (CFR: 41.10 / 45.3) (3.3/3.4)

QUESTION: 69 12209 (1 point(s))

The plant is operating at 100% power. Surveillance Procedure 6.2PCIS.302, MAIN STEAM LINE HIGH TEMPERATURE CHANNEL FUNCTIONAL TEST 9 (DIV 2) has just commenced.

STEAM TUNNEL HIGH TEMP CHANNEL A (9-5-1/E-1) then alarms (the first annunciation of this alarm for the shift).

What action(s) is/are required?

Acknowledge the annunciator . . .

a. **only**.

b. **AND** announce "expected annunciator".

c. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **only**.

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

ANSWER: 69 12209

d. **AND** announce "STEAM TUNNEL HIGH TEMP CHANNEL A" **AND** pull the associated alarm card.

This annunciator is not associated with the surveillance and therefore the alarm should be acknowledged, announced and the alarm card pulled.

<u>Answer source:</u> 2.3.1 p. 3, steps 4.12, 4.13 & 4.14

Distractors:

a. This annunciator is not expected for this surveillance.
b. This annunciator is not expected for this surveillance.
c. The alarm card must be pulled for this annunciator.

Source: *Direct from Bank*

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
70	13673	01	06/18/2002	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Procedures	INT0320104, CNS Administrative Procedures General Operating Procedures (Startup and Shutdown) Procedures (Formal Classroom/Pre-OJT Training)

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

Related Objectives	
INT032010400G0300	State from memory the " Mitigating Task Scram Actions" associated with Procedure 2.1.5, Reactor Scram.

Related References	
2.1.5	Procedure 2.1.5, Reactor Scram

Related Skills (K/A)	
295006.AA2.05	Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) Whether a reactor SCRAM has occurred (4.6*/4.6*)

QUESTION: 70 13673 (1 point(s))

The plant was operating at full power when a reactor scram occurred.

Per 2.1.5, "Reactor Scram," which methods are to be utilized to determine that all control rods have been fully inserted into the core?

a. REFUEL MODE SELECT PERMISSIVE light ON **ONLY**.

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

c. All green FULL-IN lights on the full core display are ON **AND** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

d. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** rod positions indicating 00 on the PMIS RPIS display screen.

ANSWER: 70 13673

b. The REFUEL MODE SELECT PERMISSIVE light is ON **OR** ***ALL*** green FULL-IN lights on the full core display are ON.

Answer source: 2.1.5 p. 5, Attachment 1, step 1.5

Distractors:

a. Utilizing the green full in lights on the full core display is also allowed.
c. The use of the RPIS PMIS display is not directed by procedure 2.1.5.
d. The use of the RPIS PMIS display is not directed by procedure 2.1.5

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
71	16796	01	03/20/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	5	Multiple Choice	

Topic Area	Description
Abnormal/Emergency Procedures	INT0320126, CNS Abnormal Procedures (RO) Cooling Water

Related Lessons	
INT0320126	CNS Abnormal Procedures (RO) Cooling Water

Related Objectives	
INT0320126Q0Q0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
2.4SDC	RHR Loss of Shutdown Cooling

Related Skills (K/A)	
295021.AA2.07	Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.10 / 43.5 / 45.13) Reactor recirculation flow (2.9/3.1)

QUESTION: 71 16796 (1 point(s))

With the plant shutdown in Mode 4 and RHR loop "B" operating in Shutdown Cooling, the following conditions exist:

- ➢ Reactor pressure is 0 psig
- ➢ Recirc suction temperature is 170°F
- ➢ Reactor water level is 36" (NR)
- ➢ RHR pumps "A" and "C" have both motors disconnected from their pumps

Subsequently, "B" RHR Loop develops a leak, requiring the Control Room operators to remove RHR loop "B" from service.

What action is required, and for what reason?

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

b. RPV water level must be raised to > 48" in order to maximize Reactor coolant contact with RPV metal for enhanced heat transfer to the Drywell atmosphere.

c. A Reactor Recirculation pump must be started to reduce the possibility of excessive thermal stresses on the CRD stub tubes.

d. A Reactor Recirculation pump must be started in order to reduce the possibility of thermal binding of the RHR-MO-25A/B valves caused by the expected coolant heatup.

ANSWER: 71 16796

a. RPV water level must be raised to > 48" to aid in natural circulation flow.

Answer source: 2.4SDC p. 10, Attachment 2, step 1.1

Distractors:

b. Water level is raised to enhance natural ciruclation flow, not heat transfer.
c. A Recirculation pump is started, but not to prevent thermal stresses on the stub tubes. Starting a recirc pump causes thermal stress on the stub tubes.
d. A Recirculation pump is started, but not to prevent thermal binding of the gate valves. These valves are cycled if closed during a cooldown.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
72	14426	01	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum

Related Lessons	
INT0320132	CNS Abnormal Procedures (RO) Off Gas/Vaccuum
COR0011402	OPS MAIN TURBINE

Related Objectives	
COR0011402001060D	Given a specific Main Turbine and Auxilary systems malfunction, determine the effect on any of the following: Condenser vacuum
INT0320132L0L0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).
COR0011402001140E	Given a Main Turbine and Auxiliaries component manipulation, predict and explain the changes in the following parameters: Steam seal pressure

Related References	
2.4VAC	Loss of Condenser Vacuum
2.2.75	Procedure 2.2.75, Steam Sealing System

Related Skills (K/A)	
295002.AK2.11	Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: (CFR: 41.7 / 45.8) Seal steam: Plant-Specific. (2.6/2.7)

QUESTION: 72 14426 (1 point(s))

The plant is at 30% power with the following conditions:

- ➢ Main Condenser vacuum is currently 27"Hg and degrading
- ➢ SJAE air flow (AR-FR-47) has shown a large increase over the last several minutes
- ➢ Gland Steam supply pressure indicator (MS-PI-83) indicates 0 psig

What action will correct the problem?

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

b. Throttle ***Closed*** MS-MO-BMV3, Steam Supply Bypass Valve.

c. Throttle ***Open*** MS-MO-BMV4, Steam Unloader Bypass Valve.

d. Throttle ***Closed*** MS-MO-BMV4, Steam Unloader Bypass Valve.

ANSWER: 72 14426

a. Throttle ***Open*** MS-MO-BMV3, Steam Supply Bypass Valve.

Regardless of the valve alignment or supply, the correct action to take at this power level is to open the steam bypass from main steam to put pressure on the seals.

Answer source: 2.2.75 p. 3, step 4.19.2

Distractors:

b. Taking action to close BMV3 could only make a low pressure situation worse, and BMV3 is probably already closed to begin with.
c. BMV4 would be opened if pressure was too high and steam was leaking from the seals.
d. Throttling closed BMV4 could make a high pressure situation worse, if the steam unloaders did not have enough capacity to maintain pressure.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
73	4214	00	05/21/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	INT0320134, OPS CNS Abnormal Procedures (RO) - 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).
INT0320134D0D0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).

Related References	
5.4.FIRE-SD	FIRE INDUCED SHUTDOWN FROM OUTSIDE CONTROL ROOM

Related Skills (K/A)	
2.4.34	Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications. (CFR: 43.5 / 45.13) (3.8/3.6)

QUESTION: 73 4214 (1 point(s))

A shutdown from outside the Control Room is in progress per 5.4FIRE-S/D.

How is Service Water flow established/controlled through RHR HX after the "B" and "D" Service Water Pumps have been started?

a. The Reactor Building operator manually operates SW-MO-89B from the handwheel.

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

c. The ASD Operator controls flow by the control switch for SW-MO-89B on the RHR panel in the ASD Room.

d. The Reactor Building operator fully opens SW-MO-89B at MCC-Y and the Control Building operator throttles on the SWBP discharge valve.

ANSWER: 73 4214

b. The Reactor Building operator operates the open contacts at MCC-Y for SW-MO-89B.

5.4FIRE-SD directs the operator to ensure breaker is closed and to rotate silver screw under switch which defeats mechanical interlock and open cubicle door. Then to remove control power fuse to prevent spurious operation, and to open valve by pressing button on LOWER contactor for time indicated for valve on the procedure attachment.

Answer source: 5.4FIRE-S/D p.22, step 1.1.6

Distractors:

a. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
c. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.
d. 5.4.FIRE-SD directs the use of the open and closed contacts at MCC-Y for SW-MO-89B.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
74	5127	01	03/19/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	SKL0080101, WATCHSTANDING PRINCIPLES

Related Lessons
SKL0080101 WATCHSTANDING PRINCIPLES

Related Objectives	
SKL00801010011500	State what should be done if an out of limits reading is recorded on the logs or if an error is made recording a reading.

Related References	
1.9	Control and Retention of Records

Related Skills (K/A)	
2.1.18	Ability to make accurate / clear and concise logs / records / status boards / and reports. (CFR: 45.12 / 45.13) (2.9/3.0)

QUESTION: 74 5127 (1 point(s))

The RO realizes that an incorrect river level was transcribed by the Station Operator into 6.LOG.601, DAILY SURVEILLANCE LOG - MODES 1, 2, AND 3."

How is the the wrong river level corrected?

a. Correct number should be entered **AND** dated.

b. Circle the number, **AND** write in the correct number with an explanation.

c. Circle, initial **AND** date the old number, **AND** write in the correct number.

d. Draw one line through the incorrect old number, initial **AND** date, **AND** write in the correct number.

ANSWER: 74 5127

d. Draw one line through the incorrect old number initial **AND** date, **AND** write in the correct number.

Procedure 1.9 requires a single line to be drawn through the information to be corrected such that the information crossed out must remain legible. The person making the correction must initial and date the correction. Additional information may be added.

Distractors:

a. Required to draw a single line through correction, intial & date and write in the correct entry.
b. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.
c. Action for a reading out of limit. Required to draw a single line through correction, intial & date and write in the correct entry.

<u>Source:</u> ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
75	768	1	03/10/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Administrative	SRO Only COR0012102, Refueling

Related Lessons
COR0012102 Refueling

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

Related References	
10.25	Refueling - Core Unload, Reload, and Shuffle

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

QUESTION: 75 768 (1 point(s))

A core reload is in progress.

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

Per procedure 10.25 (Refueling - Core Unload, Reload and Shuffle), what action is required to be performed?

a. Immediately terminate fuel loading.

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

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

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

ANSWER: 75 768

a. Immediately terminate fuel loading.

Answer source: 10.25 p. 12, step 8.1.18

Distractors:

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

55.43 section(s): (7)

SRO Justification: SRO licensed personnel supervise refueling activities and direct the actions of the refuel bridge operator.

Source: ***Modified***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
76	5101	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	SRO Only COR0012102, Refueling

Related Lessons	
COR0012102	Refueling
INT0320117	CNS Administrative Procedures Volume Ten, Nuclear Performance Procedures (Formal Classroom/Pre-OJT Training)

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

Related References	
10.23	New Fuel Inspection, Channeling, and Control Blade Inspection
10.25	Refueling - Core Unload, Reload, and Shuffle
2.2.31	Procedure 2.2.31, Fuel Handling - Refueling Platform

Related Skills (K/A)	
2.2.27	Knowledge of the refueling process. (CFR: 43.7 / 45.13) (2.6/3.5)

QUESTION: 76 5101 (1 point(s))

When handling NEW fuel in the Fuel Pool Area, how many bundles are allowed outside normal storage area **OR** shipping container at a time?

a. one (1)

b. two (2)

c. three (3)

d. four (4)

ANSWER: 76 5101

c. three (3)

Answer source: 10.23 p. 4, step 2.25

Distractors:

a. 3 bundles are allowed.
b. 3 bundles are allowed.
d. 3 bundles are allowed.

55.43 section(s): (7)

SRO Justification: SRO responsible for refueling and the Special Nuclear Material Executor must be an SRO licensed individual.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
77	5468	00	03/07/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Systems	SRO Only COR0020902, Digital Electro-Hydraulic Control

Related Lessons
COR0020902 Digital Electro-Hydraulic Control

Related Objectives	
COR0020902001070B	Given a specific DEH Control system malfunction, determine the effect on any of the following: Reactor pressure

Related References	
2.2.77.1	Procedure 2.2.77.1, DEH Control System
3.4.10	Reactor steam dome pressure

Related Skills (K/A)	
295007.AA2.02	Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Reactor power (4.1*/4.1*)

QUESTION: 77 5468 (1 point(s))

Reactor power is 97%. After raising pressure set on DEH, the setpoint continues to rise.

What affect (if any) will the rising pressure setpoint have on reactor power?

Reactor power will . . .

a. remain constant. Reactor pressure will remain within the steady-state Technical Specification limit.

b. lower until the minimum pressure setpoint is reached, then return to 97%. Reactor pressure will remain within the steady-state Technical Specification limit.

c. rise until the maximum pressure setpoint is reached, then return to 97%. Reactor pressure will exceed the steady-state Technical Specification limit.

d. rise until the reactor scrams on high reactor pressure or high reactor power. Reactor pressure will exceed the steady-state Technical Specification limit.

ANSWER: 77 5468

d. rise until the reactor scrams on high reactor pressure or reactor power. Reactor pressure will exceed the steady-state Technical Specification limit.

Explanation: DEH Max pressure setpoint is 1200 psig.

Answer source: Tech Spec LCO 3.4.10
2.4DEH, p 5, Attachment 1
COR002-09-02, p. 28, section 2.b

Distractors:

a. Reactor pressure will rise. The Tech Spec limit of 1020 psig will be exceeded.
b. Reactor pressure will rise. The Tech Spec limit of 1020 psig will be exceeded.
c. The reactor will scram.

55.43 section(s): (2)

SRO Justification: SRO assess plant conditions during an abnormal event and determine if the plant is responding per design. This knowledge would be used ot select the appropriate procedure to mitigate the event, as required. Knowledge of the

Tech Spec limit for Steam Dome pressure and range of DEH controls, reactor power response from memory.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
78	14052	00	05/15/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	SRO Only COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001160D	Given a Containment/PCIS component manipulation, predict and explain the changes in the following: Drywell to suppression chamber D/P
COR0020302001140F	Briefly describe the following concepts as they apply to the Primary containment: Drywell to Torus Differential Pressure.

Related References	
2.2.60	Procedure 2.2.60, Primary Containment Cooling And Nitrogen Inerting System

Related Skills (K/A)	
2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation. (CFR: 43.5 / 45.12 / 45.13) (3.7/4.4)

QUESTION: 78 14052 (1 point(s))

The plant was operating at power when drywell pressure and temperature began rising. The crew vented ***only*** the torus using the Standby Gas Treatment system. 30 minutes later, torus pressure is 0.28 psig and drywell pressure is 0.3 psig.

Is the primary containment OPERABLE and what is the basis for the conclusion?

a. Yes. The torus air space and drywell atmosphere are connected via the inerting piping, bypassing the downcomers.

b. Yes. The torus air space and drywell atmosphere are connected via the vent piping, irrespective of which air space SGT is aligned to.

c. No. A differential pressure of approximately 1.3 psid is required to overcome backpressure on the vent header downcomers if the pressure suppression function is intact.

d. No. A differential pressure of approximately 1.3 psid is required to establish flow between the the torus air space and drywell atmosphere if the primary containment is intact.

ANSWER: 78 14052

a. Yes. The torus air space and drywell atmosphere are connected via the inerting piping, bypassing the downcomers.

<u>Answer source:</u> 2.2.60 valve lineup
COR002-03-02 Figure 9a

Distractors:

b. SGT does not cross-connect the air spaces.
c. This is the expected containment response.
d. This is the expected containment response.

2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.(CFR: 43.5 / 45.12 / 45.13) as it applies to: 295010 High Drywell Pressure / 5

<u>55.43 section(s):</u> (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification assessment of containment integrity.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
79	32	0	03/13/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Integrated Plant	SRO Only COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001160B	Given a Containment/PCIS component manipulation, predict and explain the changes in the following: Drywell pressure

Related References	
5.2REC	LOSS OF REC
2.4PC	PRIMARY CONTAINMENT CONTROL

Related Skills (K/A)	
295020.AK3.03	Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: (CFR: 41.5 / 45.6) Drywell/containment temperature response (3.2/3.2)

QUESTION: 79 32 (1 point(s))

The plant has been operating at rated power for the last 6 months when the MSIVs suddenly close.

If all systems respond as designed and no operator action is taken,

1. If no operator action is taken, how does Drywell pressure respond over the next 2 hours?
2. Why does this response occur?
3. What actions will be procedurally required?

a. 1. Drywell pressure will lower slowly.
   2. The drywell heat load is significantly reduced, lowering drywell average temperature.
   3. Torus cooling will be required per EOP-3A, Primary Containment Control.

b. 1. Drywell pressure will remain relatively constant.
   2. The heat load in the drywell remains relatively constant at a given reactor pressure.
   3. Drywell FCUs must be placed in OVERRIDE per 2.4PC, Primary Containment Control.

c. 1. Drywell pressure will rise slowly.
   2. The heat added to the torus migrates to the drywell, raising average drywell temperature.
   3. Torus cooling will be required per EOP-3A, Primary Containment Control.

d. 1. Drywell pressure will rise rapidly.
   2. The loss of drywell cooling and heat addition to the torus, raising average drywell temperature.
   3. Drywell FCUs must be placed in OVERRIDE per 2.4PC, Primary Containment Control.

ANSWER: 79 32

a. 1. Drywell pressure will lower slowly.
   2. The drywell heat load is significantly reduced, lowering drywell average temperature.
   3. Torus cooling will be required per EOP-3A, Primary Containment Control.

With one recirculation pump tripped (one is powered from the Normal Transformer that trips when the Main Generator trips), and the other recirculation pump at minimum speed (< 20% feed flow limits recirc speed to 22%), the heat load on drywell cooling is significantly decreased. With RWCU isolated, a very large heat load is removed from the REC system. Drywell pressure was stable with the previous drywell heat load. The combination of these effects results in lower drywell temperature and pressure after the scram. Torus water temperature does not directly affect drywell airspace temperature.

Answer source: 2.4PC p. 3, step 5.4
5.2REC p. 2, step 4.3.3.2
5.2REC p. 5, step 5.8.2

Distractors:

b. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment. 2.4PC does not allow placing FCUs to OVERRIDE.
c. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment. Drywell pressure lowers.
d. RWCU isolates, one reactor recirc pump trips and the other reactor recirc pump goes to minimum speed, significantly reducing the heat load inside the primary containment. Drywell cooling is not lost on a normal group 1 isolation and scram. Drywell pressure lowers.

USAR XIV-5-6

55.43 section(s): (5)

SRO Justification: SRO personnel assess plant response to transients to ensure the plant response is as designed, and based on that assessment, direct plant response to the event.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
80	36	0	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	SRO Only COR0020302, CONTAINMENT

Related Lessons
COR0020302 CONTAINMENT

Related Objectives	
COR0020302001120J	Describe the Containment design features and/or interlocks that provide for the following: Secondary containment over pressure protection

Related References	
V.XII.2.3	
XII 2.3.5.2	USAR and Appendix C (loepxviii1)

Related Skills (K/A)	
295035.EK3.01	Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.5 / 45.6) Blow-out panel operation: Plant-Specific. (2.8/3.1)

QUESTION: 80 36 (1 point(s))

What is the reason for designing the Reactor Building metal siding to blow off if excessive Reactor Building to atmosphere differential pressure exists?

a. To limit the stresses on the Reactor Building during a tornado.

b. To ensure steam leaks in the steam tunnel do not migrate into the Secondary Containment quads, causing safety-related equipment to become non-functional.

c. To allow rapid restoration of Secondary Containment integrity after a large steam leak to limit releases to small fractions of 10 CFR 100 limits.

d. To ensure that any steam leakage into the Secondary Containment does not migrate to the Control Building, ensuring radioactive dose to the operators remains within analyzed limits.

ANSWER: 80 36

a. To limit the stresses on the Reactor Building during a tornado.

Answer source: USAR p. XII-2-14, section 2.3.3.2.4
USAR p. C-2-10, section 2.5.4

Distractors:

b. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.
c. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.
d. The reason for the Rx Bldg blow out panels is to limit the stresses on the Reactor Building during a tornado.

55.43 section(s): (2)

SRO Justification: SRO knowledge of basic plant design is required to assess plant response to abnormal and emergency plant events and to predict impacts of events to allow direction of appropriate actions.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
81	44	0	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Systems	SRO Only SKL0124223, OPS RESIDUAL HEAT REMOVAL

Related Lessons	
SKL0124223	OPS RESIDUAL HEAT REMOVAL
OTH0020414	Systems Precautions and Limitations Examination (SRO)

Related Objectives	
SKL012422300A0200	Explain the Residual Heat Removal system limitations and precautions as stated in the SOP 2.2.69, SOP 2.2.69.1, SOP 2.2.69.2 and SOP 2.2.69.3.
SKL012422300B0600	Comply with all related Residual Heat Removal system limits and precautions.

Related References	
2.2.69.2	Procedure 2.2.69.2, RHR System Shutdown Operations

Related Skills (K/A)	
205000.A2.12	Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate...: (CFR 41.5 / 45.6) Inadequate system flow (2.9/3.0)

QUESTION: 81 44 (1 point(s))

The plant is operating during a refueling outage with the following conditions:

- 180 fuel bundles have been removed from the reactor core and placed in the fuel pool
- "B" RHR loop is in Shutdown Cooling
- RHR-MO-27B, OUTBD INJECTION VLV is inadvertently closed for several seconds
- "B" RHR loop flow now indicates 2800 gpm

What actions are required?

a. Raise RPV water level to > +48" (NR) **only**.

b. Raise shutdown cooling flow to > 5000 gpm but ≤ 7000 gpm.

c. Shut "B" RHR loop minimum flow valve **AND** raise shutdown cooling flow to > 7000 gpm but ≤ 8400 gpm.

d. Shut "B" RHR loop minimum flow valve **AND** raise RPV water level to > +48" (NR) **AND** raise shutdown cooling flow to > 5000 gpm but ≤ 8400 gpm.

ANSWER: 81 44

b. Raise shutdown cooling flow to > 5000 gpm but ≤ 7000 gpm.

Answer source: 2.2.69.2 p. 4, step 2.26.4
2.2.69.2 p. 3, step 2.15

Distractors:

a. Reactor cavity is flooded and no loss of inventory should have occurred. Flow must be raised > 5000 gpm and ≤ 7000 gpm.
c. Flow is limited to 7000 gpm with fuel removed around instrument channels. With 180 bundles removed, at least some instrument dry tubes must be exposed.
d. Flow is limited to 7000 gpm with fuel removed around instrument channels. With 180 bundles removed, at least some instrument dry tubes must be exposed. Reactor cavity is flooded and no loss of inventory should have occurred.

55.43 section(s): (5) & (7)

SRO Justification: SRO from-memory knowledge of system precautions and limitations, prediction of integrated system response.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
82	4002	1	03/13/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070501, OPS Introduction to Technical Specifications

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070504	CNS Tech. Spec. 3.3, Instrumentation

Related Objectives	
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.
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	
3.0	"LCO and Surveillance Applicability"
3.3.1.1	Reactor protection system (RPS) instrumentation

Related Skills (K/A)	
2.1.12	Ability to apply technical specifications for a system. (CFR: 43.2 / 43.5 / 45.3) (2.9/4.0)

QUESTION: 82 4002 (1 point(s))

The plant is operating at full power with the "A" and "C" APRMs failed downscale due to internal circuitry problems. The Control Room operators have placed a half scram on RPS "A".

What effect will this condition have on the continued ability to operate the plant in Mode 1?

a. Unless one of the failed APRMs is returned to normal service prior to the next scheduled Channel Functional Test, of one of the RPS B side APRMS, the plant will have to be shutdown.

b. Continued operation in this condition is not permitted. Due to operation outside the accident analysis, LCO 3.0.3 must be entered and the plant must be placed in HOT SHUTDOWN within the next 37 hours.

c. While in this condition, the remaining operable APRMs must be channel checked every 12 hours AND channel functional tests must be performed every 7 days, with the exception that their trip units need not be tripped as long as there are no abnormal responses observed in the performance of either test.

d. Continued operation is permitted indefinitely in this condition as Tech Specs allows resetting the trip inserted due to the failed APRMs in order to perform testing required to demonstrate the operability of the operable APRMs in service that are associated with the B logic of the Reactor Protection System.

ANSWER: 82 4002

d. Continued operation is permitted indefinitely in this condition as Tech Specs allows bypassing the trip functions of the failed APRMs in order to perform testing required to demonstrate the operability of the operable APRMs in service that are associated with the B logic of the Reactor Protection System.

Answer source: Tech Spec 3.0.5

Distractors:

a. Tech spec 3.0.5 allows equipment removed from service or declared inoperable to comply with ACTIONS to be returned to service to support testing of other equipment.
b. Tech spec 3.0.5 allows equipment removed from service or declared inoperable to comply with ACTIONS to be returned to service to support testing of other equipment. Trip capability is maintained.
c. The channel functional tests are still required for "B" side instruments.

2.1.12 Ability to apply technical specifications for a system.(CFR: 43.2 / 43.5 / 45.3) as it applies to:
215005 APRM / LPRM

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification REQUIRED ACTION and application of 3.0.5.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
83	38	0	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
4	2	1	8	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070501, OPS Introduction to Technical Specifications

Related Lessons	
INT0070501	OPS Introduction to Technical Specifications
INT0070509	CNS Tech. Spec. 3.8, Electrical Power System
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.
INT00705090010300	Given a set of plant conditions that constitutes non-compliance with a Section 3.8 LCO, determine the ACTIONS that are required.
INT00705010010200	Given plant conditions and a Specification, apply the rules of Section 3.0 to determine appropriate actions.

Related References	
3.3.3.1	Post accident monitoring (PAM) instrumentation
3.8.1	AC Sources - Operating

Related Skills (K/A)	
216000.K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: (CFR: 41.7 / 45.7) A.C. electrical distribution (3.1/3.3)

QUESTION: 83 38 (1 point(s))

The plant was operating at rated power when the following occurred:

- NBI-LI-85A (Wide Range RPV water level) becomes inoperable at 0900 on 7/06
- An air leak in the starting air system for DG2 occurs at 1200 on 7/12. Air pressure lowers to 100 psig.

IF conditions do not change, which of the following is the **LATEST** that LCO 3.3.3.1 "PAM Instrumentation" allows the plant to enter **MODE 3**?

a. 1600 on 7/12.

b. 1600 on 7/17.

c. 2400 on 7/19.

d. 0400 on 7/20.

ANSWER: 83 38

d. 0400 on 7/20.

3.8.3.F requires DG2 be declared inoperable immediately.

NBI-LI-85A (Wide Range RPV water level PAM instrument powered from CCP-1A) is inoperable. DG #2 becomes inoperable requiring 4 hours later, the Conditions and Required Actions for both Wide Range RPV water level PAM instruments inoperable (3.3.3.1 Condition "C") must be entered as the inoperable PAM instrument on a division opposite that of the inoperable DG. NBI-LI-85B is powered by CCP which is supported by DG #2. NBI-LI-85A is "an inoperable redundant required feature supported by the other DG".

Enter 3.3.3.1.C at 1600 on 7/12. Enter 3.3.3.1.D at 1600 on 7/19. Enter 3.3.3.1.E at 1600 on 7/19. Be in MODE 3 0400 on 7/20

Answer source: Tech Spec LCO 3.3.3.1 and 3.8.1

Distractors:

a. Time when 3.3.3.1.C is entered.
c. Math error for adding 7 days to 7/12.
d. Time when DG spec (3.8.1) requires MODE 3

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO personnel implement Technical Specifications.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases, T.S. LCO 3.8.1 and bases, T.S. LCO 3.8.3 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
84	48	0	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070502, CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Lessons	
INT0070502	CNS Tech. Spec. 3.1, Reactivity Control Systems

Related Objectives	
INT00705020010100	Given a set of plant conditions, recognize non-compliance with a Section 3.1 LCO.
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	
3.1.3	Control rod operability
3.1.4	Control rod scram times

Related Skills (K/A)	
2.2.21	Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) (2.3/3.5)

QUESTION: 84 48 (1 point(s))

During the current refueling outage, the control rod drive mechanism was removed, rebuilt and then reinstalled for control rod 26-27.

How are control rod surveillance requirments and the plant startup coordinated?

A coupling check *must* be performed on the rod . . .

a. ***prior*** to reactor startup. Control rod scram time does not need to be tested until reactor pressure is at least 800 psig and ***must*** be completed ***prior*** to exceeding 40% power.

b. ***prior*** to reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

c. when it is fully withdrawn during the startup, but is **NOT** required ***prior*** to the reactor startup. Control rod scram time does not need to be tested until reactor pressure is at least 800 psig and ***must*** be completed ***prior*** to exceeding 40% power.

d. when it is fully withdrawn during the startup, but is **NOT** required ***prior*** to the reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

ANSWER: 84 48

b. ***prior*** to reactor startup. Control rod scram time ***must*** be tested ***prior*** to startup and tested again when reactor pressure is at least 800 psig (***prior*** to exceeding 40% power).

SR 3.1.3.5 requires coupling check prior to calling a control rod operable after work that could affect coupling. SR 3.1.4.3 (referenced from SR 3.1.3.4) requires scram testing at reduced pressure prior to calling a control rod operable after work that could affect scram time. SR 3.1.4.1 requires scram testing at or above 800 psig prior to exceeding 40% power.

Answer source: SR 3.1.3.4, SR 3.1.3.5, SR 3.1.4.1 & SR 3.1.4.3

Distractors:

a. Scram times must be checked prior to calling control rod operable.
c. Coupling check is required prior to statup. Scram times must be checked prior to calling control rod operable.
d. Coupling check is required prior to statup.

2.2.21 Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) as it applies to:
201003 Control Rod and Drive Mechanism

55.43 section(s): (2) & (6)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Integration of multiple Technical Specification surveillance requirements.

Source: ***New***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.1.3 and bases, 3.1.4 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
85	114	01	03/25/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	7	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only 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.
INT00705020010700	From memory, for MODES 1 and 2 with THERMAL POWER less than 10% RTP, state the actions required in less than one hour for nine or more OPERABLE control rods NOT in compliance with BPWS (LCO 3.1.6).

Related References	
3.1.6	Rod pattern control
3.3.2.1	Control rod block instrumentation

Related Skills (K/A)	
2.2.33	Knowledge of control rod programming. (CFR: 43.6) (2.5/2.9)

QUESTION: 85 114 (1 point(s))

A plant casualty occurred which required a rapid power reduction to 8% RTP. As a result, the RWM has initiated control rod Insert and Withdrawal blocks. Seventeen (17) control rods are Withdrawal errors and two (2) control rods are Insert errors.

What actions are required per Technical Specifications?

a. Suspend withdrawal of control rods immediately. Place the reactor mode switch in the shutdown position within 1 hour. RWM may be bypassed and actions may continue to insert Withdrawal error control rods.

b. Bypass RWM and insert/withdraw the control rods to the correct position per the Banked Position Withdrawal Sequence (BPWS) OR declare associated control rods inoperable with 8 hours. Movement of control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

c. Bypass RWM and withdraw the Insert error control rods to the correct position per the Banked Position Withdrawal Sequence (BPWS). Declare the Withdrawal error control rods inoperable with 8 hours. Movement of the control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

d. Suspend control rod movement except by scram immediately OR verify at least 12 rods are still withdrawn immediately OR verify that startup with RWM inoperable has not been performed within the last calendar year immediately. Movement of control rods shall be verified in accordance with BPWS by a second licensed operator or other qualified member of the technical staff.

ANSWER: 85 114

a. Suspend withdrawal of control rods immediately. Place the reactor mode switch in the shutdown position within 1 hour. RWM may be bypassed and actions may continue to insert Withdrawal error control rods.

Answer source: Tech Spec 3.1.6 and 3.3.2.1

Distractors:

b. Must discontinue withdrawal of control rods. RMS must be placed in shutdown within 1 hour.
c. Must discontinue withdrawal of control rods. RMS must be placed in shutdown within 1 hour.
d. Can insert rods. The 12 rod specification is specific to RWM operability and plant startup only.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO ability to implement Tech Specs.

Source: ***Direct from bank***

Provide to Candidate: T.S. 3.0 section and bases, 3.3.2.1 and bases.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
86	42	0	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Integrated Plant	SRO Only INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)

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

Related Objectives	
INT00705050010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.4 Specification.

Related References	
3.4.2	Jet pumps

Related Skills (K/A)	
202001.K1.06	Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Jet pumps (3.6/3.6)

QUESTION: 86 42 (1 point(s))

Why is a plant shutdown required if one or more jet pumps is determined to be inoperable?

a. Excessive vibration of the jet pumps may occur.

b. Increased risk of uncontrolled thermal hydraulic oscillations.

c. The assumed blowdown flow during a LOCA may exceed the analyzed value.

d. The reactor may not have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

ANSWER: 86 42

c. The assumed blowdown flow during a LOCA may exceed the analyzed value.

Answer source: Tech Spec Bases p. B 3.4-10, top paragraph

Distractors:

a. Basis for recirc pump speed mismatch.
b. Basis for power/flow limitations.
d. Basis for loop flow mismatch.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO knowledge of Tech Spec basis.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
87	16415	01	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Systems

Related Lessons	
INT0320101	CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT032010100G010L	Discuss the following as described in Administrative Procedure 0.26, Surveillance Program: Precautions and Limitations
INT00705070010100	Given a set of plant condtions, recognize non-compliance with a Chapter 3.6 LCO.

Related References	
3.6.1.3	Primary Containment Isolation Valves (PCIVs)
3.6.2.3	Residual Heat Removal (RHR) Suppression Pool Cooling

Related Skills (K/A)	
2.2.21	Knowledge of pre and post maintenance operability requirements.(CFR: 43.2) (2.3/3.5)

QUESTION: 87 16415 (1 point(s))

The plant is at power with the 6.1RHR.201, RHR Power Operated Valve Operability Test in progress. When attempting to time open RHR-MO-34A, SUPPR POOL COOLING INBD THROTTLE VLV, the valve remained closed and the breaker tripped at the MCC. The Station Operator reports an acrid smell at the MCC. The valve motor actuator gear set was replaced last outage.

Which one of the following describes . . .

(1) if the breaker can be reset to try to stroke the valve (yes or no)
(2) if the breaker will not reset, are TS 3.6.1.3 and/or TS 3.6.2.3 entered?

a. (1) No (2) Both TS are entered

b. (1) No (2) Only TS 3.6.2.3 is entered

c. (1) Yes (2) Both TS are entered

d. (1) Yes (2) Only TS 3.6.2.3 is entered

ANSWER: 87 16415

a. (1) No (2) Both TS are entered

Immediately declare the valve inoperable and enter the ACTIONS of TS 3.6.2.3. TS 3.6.1.3 is also entered because the valve is a primary containment isolation valve.

Answer source: Tech Spec 3.6.2.3 and 3.6.1.3

Distractors:

b. TS 3.6.1.3. entry is required because this is a PCIS valve.
c. Breaker cannot be reset and closed.
d. Breaker cannot be reset and closed. TS 3.6.1.3. entry is required because this is a PCIS valve

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO responsible for sureillance requirements and implementing requirements.

Source: ***Direct from bank***

Provide to Candidate: T.S. 3.0 section and bases, T.S. LCO 3.6.1.3 and bases, 3.6.2.3 and bases, 6.1RHR.201**.**

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
88	24	0	03/07/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Systems

Related Lessons	
INT0070507	CNS Tech. Spec. 3.6, Containment Systems

Related Objectives	
INT00705070010200	Discuss the applicable Safety Analysis in the Bases associated with each Chapter 3.6 specification.

Related References	
3.6.2.1	Suppression pool average temperature

Related Skills (K/A)	
295013.AK1.04	Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.8 to 41.10) Complete condensation. (2.9/3.2)

QUESTION: 88 24 (1 point(s))

The plant is operating at rated power with the following conditions:

- ➢ High Pressure Coolant Injection (HPCI) was in service for the quarterly full-flow test.
- ➢ The crew removed HPCI from service when average torus water temperature reached 105°F.
- ➢ 26 hours after HPCI was secured, average torus water temperature is 98°F (due to elevated Missouri River temperature.)

What is a potential consequence of continuing to operate the reactor at power under these conditions?

a. HPCI will not be available during a LOCA due to elevated lube oil temperature.

b. The capacity of the torus-to-drywell vacuum breakers will be exceeded during a LOCA.

c. The reactor building-to-torus vacuum breakers will operate if drywell sprays are initiated during a LOCA.

d. Steam will not completely condense at the outlet of the drywell to suppression pool downcomers during a LOCA.

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ANSWER: 88 24

d. Steam will not completely condense at the outlet of the drywell to suppression pool downcomers during a LOCA.

Answer source: Tech Spec bases p. B 3.6-51

Distractors:

a. HPCI is normally aligned to the ECST and lube oil is not affected until 140°F.
b. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.
c. This would be an issue for initiating drywell spray with evaporative cooling and is positively affected by high torus temp, not negatively affected.

55.43 section(s): (2)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. Technical Specification bases for maximum suppression pool temperature from memory.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
89	40	0	03/13/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070509, CNS Tech. Spec. 3.8, Electrical Power System

Related Lessons	
COR0020802	DIESEL GENERATORS
INT0070509	CNS Tech. Spec. 3.8, Electrical Power System

Related Objectives	
COR0020802001090C	Describe the Diesel Generator design feature(s) and/or interlock(s) that provide for the following: Speed Droop Control
INT00705090010100	Given a set of plant condtions, recognize non-compliance with a Section 3.8 LCO.

Related References	
2.2.20	Procedure 2.2.20, Standby AC Power System (Diesel Generator)
2.2.20.1	Procedure 2.2.20.1, Diesel Generator Operations

Related Skills (K/A)	
264000.K5.04	Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): (CFR: 41.5 / 45.3) Governor control (2.4/2.5)

QUESTION: 89 40 (1 point(s))

How is DG1 operation affected if the DROOP/PARALLEL Switch left in DROOP?

During a design bases LOCA coincident with off-site power, DG1 will . . .

a. reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

b. **NOT** reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

c. reach rated voltage and frequency within design time limits. Bus voltage and frequency will **NOT** be maintained as assumed in the accident analysis as bus loads sequence on.

d. **NOT** reach rated voltage and frequency within design time limits. Bus voltage and frequency will **NOT** be maintained as assumed in the accident analysis as bus loads sequence on.

ANSWER: 89 40

a. reach rated voltage and frequency within design time limits. Bus voltage and frequency will be maintained as assumed in the accident analysis as bus loads sequence on.

Answer source: 2.2.20 p. 23, step 1.4.3.6

Distractors:

b. The droop mode does not affect start time or sequence.
c. The DG will shift to Isochronous mode automatically when an auto start signal exists. This overrides the switch position on the panel for droop.
d. The droop mode does not affect start time or sequence. The DG will shift to Isochronous mode automatically when an auto start signal exists. This overrides the switch position on the panel for droop.

55.43 section(s): (5)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Specification and action required for non-compliance. SRO knowledge of diesel logic and plant accident analysis.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
90	46	0	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Technical Specifications, ODAM, TRM	SRO Only INT0070604, TRM - Fire Protection

Related Lessons	
INT0070604	TRM - Fire Protection

Related Objectives	
INT0070604001010D	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.4 High Pressure Carbon Dioxide Extinguishing Systems
INT0070604001010F	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.6 Fire Hose Stations
INT0070604001030B	Given plant conditions and the TRM, determine the ACTIONS required per the following TLCOs: T.3.11.2 Fire Suppression Water System
INT0070604001010B	Given plant conditions, determine if the following TRM Limiting Conditions for Operation (TLCOs) are met: T.3.11.2 Fire Suppression Water System

Related References	
2.2.2	Procedure 2.2.2, Carbon Dioxide Systems
T3.11.6	Fire Hose Stations

Related Skills (K/A)	
286000.K6.02	Knowledge of the effect that a loss or malfunction of the following will have on the FIRE PROTECTION SYSTEM (CFR: 41.7 / 45.7) D. C . electrical distribution (2.8*/2.9*)

QUESTION: 90 46 (1 point(s))

The plant is operating at rated power when 125 VDC is lost to the CO_2 hose station subsystem.

What is the effect on the CO_2 hose stations and what are the **MINIMUM** actions required?

On loss of DC power to Cardox System, hose stations will . . .
(NOTE: Choices are listed in MINIMUM to MAXIMUM order.)

a. charge but not vent. Document on Fire Impairment Form that no compensatory actions required.

b. vent but not charge. Contact Fire Protection Group for determination of appropriate compensatory measure based on nature of impairment and its impact on safe shutdown timeliness.

c. vent but not charge. Establish 2 hour Fire Watch patrol in affected areas within 2 hours.

d. charge but not vent. Establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within one (1) hour AND restore to OPERABLE status within 14 days.

ANSWER: 90 46

a. charge but not vent. Document on Fire Impairment Form that no compensatory actions required.

Answer source: 2.2.2 p.2, step 2.3
0.23 p. 13, Attachment 1

Distractors:

b. This is action for Non-Tech Spec Appendix R Safe Shutdown Capability. Fire hose stations are not Appendix R.
c. 0.23 required action for non-TRM fire detection equipment. This is not fire detection equipment.
d. TRM action for inoperable required fire detection equipment. This is not a TRM system and is not fire detection equipment.

55.43 section(s): (2) & (5)

SRO Justification: SRO persons assess plant conditions and determine compliance with Technical Requirements Manual (TRM) and action required for non-compliance. SRO use of TRM, administrative procedures and fire impairments.

Source: ***New***

Provide to Candidate: TRM section 3.11 and 0.23 "CNS FIRE PROTECTION PLAN".

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
91	47	0	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Off-Site Dose Assessment Manual	SRO Only INT0070702, ODAM Specifications

Related Lessons
INT0070702 ODAM Specifications

Related Objectives	
INT00707020010200	Given the ODAM Appendix D and given plant condition, determine whether there is non-compliance with the ODAM Specification Sections 3.1 thru 3.5.

Related References	
D3.2.7	Primary Containment Venting and Purging

Related Skills (K/A)	
2.3.9	Knowledge of the process for performing a containment purge. (CFR: 43.4 / 45.10) (2.5/3.4)

QUESTION: 91 47 (1 point(s))

The plant has been shutdown for 30 days for a refueling outage. A plant startup is in progress with reactor power currently 25%. Preparations are underway to purge the primary containment with nitrogen.

When the primary containment is purged, should the purge flowpath utilize Standby Gas Treatment and why/why not?

a. Yes. The SBGT system shall be used to vent or purge the primary containment to ensure deflagration limits are not exceeded at the Elevated Release Point (ERP).

b. Yes. The SBGT system shall be used to vent or purge the primary containment while coolant temperature is greater than 200°F to ensure dose rates do not exceed small fractions of 10 CFR 100 limits.

c. No. To minimize the time that Oxygen concentration is above 4% to reduce the risk of a combustible atmosphere if a LOCA were to occur, the large purge and vent valves should be used.

d. No. To minimize the time that SBGT system is on line while coolant temperature is greater than 200°F to reduce the risk of damage to SBGT system if a LOCA were to occur with the main purge and vent valves open.

ANSWER: 91 47

d. No. To minimize the time that SBGT system is on line while coolant temperature is greater than 200°F to reduce the risk of damage to SBGT system if a LOCA were to occur with the main purge and vent valves open.

Answer source: ODAM bases p. 3.2-7

Distractors:

a. The large vent/purge valves should be used under these conditions.
b. The large vent/purge valves should be used under these conditions.
c. The basis for using the large valves is to minimize risk to SBGT.

55.43 section(s): (2) & (4)

SRO Justification: SRO persons assess plant conditions and determine compliance with Off site Dose Assessment Manual (ODAM) and action required for non-compliance.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
92	30	0	03/12/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Emergency Operating Procedures	SRO Only INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Lessons	
INT0080613	OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL

Related Objectives	
INT00806130010900	Explain why HPCI but not RCIC must be secured at a primary containment water level of 11 feet.
INT00806130011100	Given plant conditions and EOP Flowchart 3A, PRIMARY CONTAINMENT CONTROL, determine required actions.

Related References	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure

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: 92 30 (1 point(s))

A LOCA has occurred with the following conditions:

- HPCI is injecting at 3000 gpm
- NO other high pressure injection source is available
- Reactor pressure is being maintained 800 to 1000 psig
- RPV water level is -102 inches (stable WR)
- HPCI suction is aligned to the torus
- Primary containment water level is 11.2 feet **AND** is dropping 6"/minute
- Average torus water temperature is 135°F **AND** is rising

What actions are required regarding HPCI operation?

a. Defeat HPCI high temperature interlocks.

b. Trip HPCI and place HPCI auxiliary oil pump in Pull-to-Lock.

c. Increase HPCI flow to restore and maintain RPV water level +3" to +54".

d. Reduce HPCI flow as necessary to maintain HPCI oil temperatures below 140°F.

ANSWER: 92 30

b. Trip HPCI and place HPCI auxiliary oil pump in Pull-to-Lock.

Answer source: EOP flowchart 3A, step SP/L-10

Distractors:

a. HPCI must be placed in PTL per EOP-3A. There is no need to defeat HPCI high temperature isolations.
c. HPCI must be placed in PTL irrespective of adequate core cooling.
d. The 140°F caution is for torus water temperature, not oil temperatures.

55.43 section(s): (5)

SRO Justification: SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

Source: ***New***

Provide to Candidate: EOP-3A with cautions and entry conditions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
93	34	0	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
2	3	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	SRO Only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RADIOACTIVITY RELEASE CONTROL

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

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

Related References	
EOP 5A, SCCP	EOP 5A, Secondary Containment Control

Related Skills (K/A)	
295032.EA2.01	Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Area temperature (3.8*/3.8)

QUESTION: 93 34 (1 point(s))

The plant was operating at rated power when a steam leak on RCIC occurred. The crew attempted to isolate RCIC with no success. The reactor has been scrammed and the following conditions now exist:

- The Reactor Building is inaccessible due to the steam leak
- A complete loss of pneumatics (air and nitrogen) to the Reactor Building exists
- Reactor pressure is being maintained at 926 psig by DEH
- Reactor water level is at 25" (NR) and stable
- NE Quad temperature is 205°F (rising)
- SE Quad temperature is 170°F (rising)
- NW Quad temperature is 200°F (rising)
- SW Quad temperature is 165°F (rising)
- RWCU area temperature is 135°F (rising)
- 903' area temperature is 135°F (rising)

What action is required by the EOPs?

a. Emergency Depressurize the RPV.

b. Cooldown the RPV at < 100 °F/hr.

c. Place SRV control switches in AUTO.

d. Fully open main turbine bypass valves.

ANSWER: 93 34

a. Emergency Depressurize the RPV.

Answer source: EOP flowchart 5A, step SC-14

Distractors:

b. Emergency Depressurization is required when 2 areas exceed maximum safe operating value.
c. Emergency Depressurization is required when 2 areas exceed maximum safe operating value. 5.8.1 says to put the SRV switches in AUTO when there has been a loss of pneumatics, but the ED guidance supercedes.
d. Emergency Depressurization is required when 2 areas exceed maximum safe operating value. If "Emergency Depressurization is Required", then "Emergency Depressurization is NOT anticipated"

55.43 section(s): (5)

SRO Justification: SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

Source: ***New***

Provide to Candidate: EOP-1A and EOP-5A with entry conditions and cautions removed.

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
94	16569	01	03/25/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	N Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	2	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	SRO Only INT0080618, EOP AND SAG GRAPHS

Related Lessons
INT0080618 EOP AND SAG GRAPHS

Related Objectives	
INT00806180010300	Given plant conditions and the EOP and SAG Graphs Flowchart, determine if operation is within the allowed region of a graph.
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	
EOP 3A, PCCP	EOP 3A Primary Containment Control Procedure
5.8	Procedure 5.8, Emergency Operating Procedures (EOPs)

Related Skills (K/A)	
295026.EA1.01	Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.7 / 45.6) Suppression pool cooling (4.1/4.1)

QUESTION: 94 16569 (1 point(s))

A LOCA has occurred with the following conditions:

- ➢ PMIS is unavailable.
- ➢ RHR pump "C" in the suppression pool cooling, torus spray and drywell spray mode.
- ➢ Torus pressure is 4 psig (stable)
- ➢ Torus average water temp is 185°F (rising slowly)
- ➢ Primary containment water level is 7 feet (stable)

The SS directs that all EOP Cautions be complied with.

Which of the following is the HIGHEST "C" RHR pump flow allowed?

a. 0 gpm

b. 5,000 gpm

c. 6,500 gpm

d. 9,000 gpm

ANSWER: 94 16569

c. 6,500 gpm.

With 4 psig overpressure and 3 feet of water above the suctions there is 5.29 psig overpressure requiring that flow be reduced to no more than 7000 gpm for NPSH concerns. Flow is also in the unsafe region of the vortex limit curve, but this curve is less limiting than the NPSH curve for this case.

Answer source: EOP graph 5 and NOTE 3

Distractors:

a. Correct if wrong (0#) curve is used.
b. Correct if wrong curve (CS) is used.
d. Correct if wrong 5# curve is followed to end of graph.

55.43 section(s): (5)

SRO Justification: SRO licensed personnel must assess NPSH requirement if PMIS is unavailable. SRO licensed personnel assess plant conditions during implementation of the EOPs and, based on that assessment, direct EOP actions.

Source: ***Direct from bank***

Provide to Candidate: **All EOP graphs.**

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
95	28	0	03/07/2003	05/21/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	SRO Only INT0320123, CNS Abnormal Procedures (RO) Reactivity

Related Lessons	
INT0320123	CNS Abnormal Procedures (RO) Reactivity

Related Objectives	
INT0320123F0F0100	Given plant condition(s), determine from memory the appropriate Abnormal/Emergency Procedure(s) to be utilized to mitigate the event(s).
INT0320123G0G0100	Given plant condition(s), determine from memory all immediate operator actions required to mitigate the event(s).

Related References	
2.4RXPWR	Reactor Power Anomalies

Related Skills (K/A)	
295014.AA2.03	Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: (CFR: 41.10 / 43.5 / 45.13) Cause of reactivity addition (4.0/4.3*)

QUESTION: 95 28 (1 point(s))

Reactor power is 20% with rods being pulled in preparation to sychronize the generator. Suddenly, Reactor power increases rapidly to 23% and stabilizes. There was no change in core plate differential pressure or reactor pressure ***prior*** to the power rise.

What caused the reactor power increase ***AND*** what procedure contains actions to mitigate this condition?

a. Dropped control rod
2.4CRD, CRD Trouble

b. Dropped control rod
2.4RXPWR, Reactor Power Anomalies

c. Loss of feedwater heating
2.4EX-STM, Extraction Steam Abnormal

d. Loss of feedwater heating
2.4MC-RF, Condensate and Feedwater Abnormal

ANSWER: 95 28

b. Dropped control rod
2.4RXPWR, Reactor Power Anomalies

Answer source: 2.4RXPWR pp. 2 & 3
Reactor Theory

Distractors:

a. 2.4CRD does not address a dropped control rod.
c. There is no feedwater heating prior to loading the turbine.
d. There is no feedwater heating prior to loading the turbine

55.43 section(s): (5)

SRO Justification: Prediction of plant response and application with discrimination necessary to select the correct procedure to mitigate the event.

Source: ***New***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
96	5668	04	03/18/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	SRO Only INT0320134, OPS CNS Abnormal Procedures (RO) - Fire

Related Lessons	
INT0320102	CNS Administrative Procedures Site Services Procedures (Formal Classroom/Pre-OJT Training)

Related Objectives	
INT032010200C030A	Discuss the requirement associated with the following items: Visitor acces and departure

Related References	
5.4FIRE	General Fire Procedure
1.1	Station Security

Related Skills (K/A)	
2.4.25	Knowledge of fire protection procedures. (CFR: 41.10 / 45.13) (2.9/3.4)

QUESTION: 96 5668 (1 point(s))

The plant is operating at rated power at 0210 on a Tuesday morning when a fire started in the reactor building. The following conditions exist:

- ➢ The fire has been burning out of control for 20 minutes
- ➢ The fire brigade has been dispatched and is attempting to contain the fire
- ➢ The fire brigade leader has requested assistance from the Nemaha Fire Department
- ➢ The plant has been scrammed
- ➢ The Nemaha Fire Department has arrived at the plant

Which of the following **SHALL** occur to allow the members of the Nemaha Fire Department to access the plant?

a. Drive on site as soon as Security opens the gate.

b. They must sign the Visitor's Log, and have their vehicles searched.

c. Access is authorized by the Security Shift Supervisor, once vehicle search and headcount are complete.

d. They must obtain an emergency TLD from security, and have the Shift Supervisor/Security Shift Supervisor authorize vehicle access.

ANSWER: 96 5668

d. They must obtain an emergency TLD from security, and have the Shift Supervisor/Security Shift Supervisor authorize vehicle access.

Answer source: 1.1 p. 3, step 3.2.2
1.1 p. 3, step 4.1
1.1 p. 6, step 5.2.3

Distractors:

a. Security must issue TLDs and either the Security Shift Supervisor or the Shift Supervisor must authorize access.
b. Vehicle search is not required and the signing the visitors log is not required.
c. Vehicle search is not required prior to access.

2.4.25 Knowledge of fire protection procedures. (CFR: 41.10 / 45.13) as it applies to 600000 Plant Fire on Site

55.43 section(s): (5)

SRO Justification: Shift Supervisor is one of the persons that can authorize access under these conditions (RO licensed personnel cannot) .

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
97	9015	01	03/07/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	Y Y N

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

Topic Area	Description
Abnormal/Emergency Procedures	SRO Only INT0320136, CNS Abnormal Procedures (RO) Miscellaneous

Related Lessons	
INT0070505	CNS Tech. Spec. 3.4, Reactor Coolant System (RCS)
INT0320136	CNS Abnormal Procedures (RO) Miscellaneous

Related Objectives	
INT00705050010800	From memory, in MODES 1, 2, or 3, state the actions required in less than one hour if RCS pressure and temperature (P/T) limits LCO is not met (LCO 3.4.9).
INT0320136P0P0100	Given plant condition(s), and the applicable Abnormal/Emergency Procedure, determine the correct subsequent actions required to mitigate the event(s).

Related References	
3.4.9	RCS Pressure and Temperature (P/T) Limits
5.1ASD	Shutdown From Outside The Control Room

Related Skills (K/A)	
295016.AA2.06	Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.10 / 43.5 / 45.13) Cooldown rate. (3.3/3.5)

QUESTION: 97 9015 (1 point(s))

Control room abandonment is required due to toxic gas in the control room. The CRS at the Alternate Shutdown Panel commences a reactor cooldown. The CRS lowers reactor pressure from 900 psig to 550 psig in 30 minutes.

If the cooldown continues at the SAME RATE (°F/minute) for the next thirty minutes, what will be the status of cooldown rate limits?

a. No limits will be exceeded.

b. Administrative limits will be exceeded; but no technical specifications will be violated.

c. Administrative limits and technical specifications will be exceeded; thirty minutes is the maximum time allowed to restore cooldown limits.

d. Administrative limits and technical specifications will be exceeded; one hour is the maximum time allowed to restore cooldown limits.

ANSWER: 97 9015

c. Administrative limits and technical specifications will be exceeded; thirty minutes is the maximum time allowed to restore cooldown limits.

Cooldown rate is 110ºF/hour (900 psig + 14.7 = 915 psia ----> 534ºF,
550 psig + 14.7 = 565 psia ----> 479ºF, 534ºF - 479ºF = 55ºF / 30 minutes or 110ºF/hr)

Answer source: Steam tables
Tech Spec 3.4.9

Distractors:

a. is incorrect; administrative limit is <or= 90ºF and is being exceeded.
b. is incorrect; technicial specification limit is <or= 100ºF and is being exceeded.
d. is incorrect; technical specifications only allows thirty minutes to restore cooldown rate.

55.43 section(s): (2) & (5)

SRO Justification: SRO licensed personnel perform actions at the Alternate Shutdown Panel and performs the cooldown from outside control room.

Source: ***Direct from Bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
98	5760	05	03/25/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	SRO Only INT0320101, CNS Administrative Procedures Volume 0, Administrative Procedures (Formal Classroom/Pre-OJT Training)

Related Lessons	
SKL0080102	OPS WATCHSTANDING PRINCIPLES FOR LICENSED OPERATORS

Related Objectives	
SKL00801020011300	State the proper response to an identified procedural deficiency which adversely affects the performance of the procedure.
SKL00801020011400	Describe the instant, non-intent procedure change process and the conditions under which it may be used.

Related References	
0.1	Introduction to CNS Operations Manual
0.26	Surveillance Program
0.4	Procedure Change Process

Related Skills (K/A)	
2.2.6	Knowledge of the process for making changes in procedures as described in the safety analysis report. (CFR: 43.3 / 45.13) (2.3/3.3)

QUESTION: 98 5760 (1 point(s))

Post Maintenance Testing is required to be performed by IAC on a pressure switch. The PMT requires the use of a portion of a surveillance procedure. The testing requirements cannot be met with the procedure as written. The procedure can be changed without changing the intent of the procedure, and the change must be made instantly due to pending work stoppage.

Who is responsible for approving the procedure change so the PMT may be completed?

a. Any two individuals holding an SRO license.

b. IAC Supervisor and the duty Shift Supervisor.

c. The on-shift Shift Supervisor with SORC concurrence.

d. The on-shift Shift Supervisor and a second SRO licensed individual.

ANSWER: 98 5760

d. The on-shift Shift Supervisor and a second SRO licensed individual.

Answer source: 0.4 p. 3, step 5.5

Distractors:

a. One must be on-shift SS.
b. IAC Supervisor not allowed.
c. SORC concurrence not required.

55.43 section(s): (3)

SRO Justification: SRO knowledge of instant procedure change requirements. Only SRO licensed personnel are allowed to make instant procedure changes.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
99	12218	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	SRO Only INT0320116, Health Physics Procedures

Related Lessons	
INT0320116	Health Physics Procedures

Related Objectives	
INT03201160000200	State the requirements an individual must meet in order to exceed any whole body exposure limits

Related References	
9.ALARA.1	Procedure 9.ALARA.1, Personnel Dosimetry Program
0.ALARA.7	Planned Special Exposure

Related Skills (K/A)	
2.3.4	Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) (2.5/3.1)

QUESTION: 99 12218 (1 point(s))

During the current calendar year a Station Operator had received a Planned Special Exposure (PSE) of 3.5 rem (TEDE) and an Occupational Exposure of 0.5 rem (TEDE).

What is the MAXIMUM dose (if any) this Station Operator can receive during the remainder of the calendar year ***without*** obtaining any additional written approval?

a. 0.0 rem

b. 0.5 rem

c. 1.5 rem

d. 2.5 rem

ANSWER: 99 12218

b. 0.5 rem

The PSE is separate from the allowed calendar year exposure. Special permission is not required as long as the calendar year exposure remains below a total of 1000 mrem (1 rem). Since the calendar year exposure is currently 0.5 rem any additional exposure that brings the total to 1.0 rem requires written approval.

Answer source: 9.ALARA.1, p. 12, Section 6.1
0.ALARA.7, p. 9, step 1.1

Distractors:

a can receive another 0.5 rem without approval.
c,d. is incorrect. Because written approval must be obtained before exceeding 1000 mrem.

55.43 section(s): (4)

SRO Justification: Authorization of exposure in excess of adminiatrative limits is a supervisory function. SRO licensed personnel are supervisors of ROs and SOs. SRO knowledge of PSE and TEDE requirements is required to assess and authorize extensions of a limit.

Source: ***Direct from bank***

Question Number	Question ID	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
100	9684	01	03/17/2003	06/13/2003	NRC Style Question	RO: SRO: NLO:	N Y N

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

Topic Area	Description
Administrative	SRO Only SKL0110101, SRO Upgrade Self-Study Program

Related Lessons	
INT0231001	Shutdown Risk Management
SKL0110101	SRO Upgrade Self-Study Program

Related Objectives	
SKL0110101001350A	0.50, Outage Management Program: Define the following terms as they apply to Administrative Procedure 0.50, Outage Management Program: 1) Acceptable isolation barrier 2) Available 3) Contingency Plan 4) Decay Heat Removal (DHR) capability 5) Defense in Depth 6) Functional 7) Key Safety Function 8) Operations with Potential for Draining the Reactor Vessel (OPDRV's) 9) Primary System Boundary 10) Protected.

Related References	
0.50	Outage Management Program

Related Skills (K/A)	
2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control. (CFR: 43.5 / 45.12) (3.7/4.3)

QUESTION: 100 9684 (1 point(s))

The plant is in day 9 of a scheduled refueling outage. The core off load has been delayed and the cavity has not yet been flooded. RHR loop A is operating in shutdown cooling with RHR pump A. RHR pump C is out of service due to an electrical fault that developed on the pump motor breaker during the previous shift. The following maintenance activities are in progress:

- RHR pumps "B" and "D" impeller replacement
- #2 Diesel Generator Main Bearing replacement
- Core Spray "B" is out of service for pump seal work
- RWCU is out of service to replace the pump seals
- Steam Separator lift

Which of the following activities would restore plant configuration to within guidelines?

a. Restore RWCU to service.

b. Restore "B" Core Spray to service.

c. Restore #2 Diesel Generator to service.

d. Flood the reactor cavity and remove the fuel pool gates.

ANSWER: 100 9684

d. Flood the reactor cavity and remove the fuel pool gates.

The current plant configuration does not provide a backup means of decay heat removal. By flooding the reactor cavity a recognized mode of backup decay heat removal is established.

Answer source: 0.50 p. 51, Attachment 4, Backup Systems.
0.50 p. 51, Attachment 4, RWCU item.
0.50 p. 51, Attachment 4, Fuel Pool/Gates item.

Distractors:

a. The RWCU system is not capable of providing backup decay heat removal at this time.
b. Core Spray does not provide decay heat removal.
c. Restoring the DG to service would not provide a mode of decay heat removal.

NOTE: No decay heat curves needed for this question. It is expected the operator should know that RWCU will not provide adequate decay heat removal 9 days after shutdown.

55.43 section(s): (6)

SRO Justification: SRO personnel authorize work during outages and assess risk and defense in depth continuously. SROs direct activities to restore the plant configuration to within the guidelines.

Source: ***Direct from bank***

Provide to Candidate: 0.50, Outage Management Program.

Question Number	Record Number	Pass %	Comment	Recommendation
RO/SRO 1	1111	50	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to identify which breakers trip if excessive amps flow when the Diesel is in parallel with the grid. The student text (COR001-01-01 and COR002-08-02) adequately address the breaker trips and interlocks.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO/SRO 4	14035	25	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to determine the effect on VARs and field amps when underexcited and grid voltage increases.	Revise student text (COR001-13-01) to address this condition. (Notification 10254366) Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO/SRO 13	14040	50/88	The test item had the operator predict how core spray would respond to an initiation signal while in full flow test and above the shutoff head for Core Spray. The test item did not specify a specific time frame to assess the Core Spray parameters. As the full-flow test valve is very fast and the minimum flow valve is very slow, the initial response will be to approach shutoff head, then lower to a discharge pressure slightly lower than shutoff head. After discussion with chief examiner, will also accept "b". See formal submittal for details.	Modify the test item to specify evaluation after valves have completed repositioning for existing conditions. (Notification 10254338) Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO/SRO 21	5425	13/100	The test item choices are confusing as choice "a" can be read as there are no inputs that currently have AC power. Choice "b" is confusing in that it states irrespective of reactor pressure when reactor pressure must exceed 1080 psig once for LLS to arm. The stem states no operator action will be taken, but to arm irrespective of reactor pressure, the control switch must be manipulated. The test item required the candidate to determine the effect a loss of RPS has on the operation of LLS logic. After discussion with chief examiner, will also accept "a". See formal submittal for details.	Reword choice "a" to "all relays providing input into the LLS logic are DC powered." Reword choice "b" to remove "irrespective of reactor pressure" (Notification 10254489) Revise student text (COR002-16-02) to address how RPS logic and power interface with LLS, including the effect a loss of RPS power has on LLS logic. (Notification 10254490) Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO/SRO 24	14451	50	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to determine the effect a failed flow transmitter will have on RCIC operation. The student text explicitly covers this failure and its effects.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO/SRO 25	14027	50	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to calculate a temperature change and then apply the temperature change to a pressure change.	Consider adding a fundamentals review module to initial license class to address the theory items in 55.41. (Notification 10254491) Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO/SRO 36	14025	50	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know if the Core Spray break detection sensed above or below core plate pressure. The SLC student text erroneously identifies the sparger as the input into the Core Spray break detection (below core plate).	Revise SLC student text to correctly describe the connections and cause/effect relationship for the Core Spray break detection (COR002-29-02) (Notification 10254492) Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO/SRO 41	8970	25	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the bases of the MCPR safety limit from memory.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO/SRO 54	54	38	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the sequence of steps to stop and prevent feedwater injection from memory. This was a recent procedure change. All incorrect choices included placing the Feedwater Level controllers in MDEM vice the required MDVP.	Cover this information during NRC post-exam review for current license class. Include why MDVP mode is used and why MDEM is not. (Notification 10254273)
RO/SRO 67	4243	50	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to calculate and apply the 25% extension allowed for Tech Spec surveillances. The students were provided with Tech Spec SR 3.0.2 which supports this concept.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO 79	14041	33	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the effect that temperature has on radionuclide adsorption. The training material adequately addresses this issue.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO 87	14034	0	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the effect that a loss of DC power has on a running DG. The training material adequately addresses this issue.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO 89	1238	33	The stem should have specified “reset and remain reset” The test item required the candidate to know how to reset a control rod drift indication.	Modify the question to specify “reset and remain reset” (Notification 10254493) Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
RO 92	5401	33	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the effect that a loss of RPS has on RBM with an edge control rod selected. The student had to determine the selected rod was an edge rod.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
RO 100	12222	0	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the minimum actions required when changing RWPs while staying in the RCA.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
SRO 76	5101	0	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the maximum number of bundles allowed outside a shipping container and the normal storage area.	Consider improving the objectives and training material pertaining to refueling administrative requirements. (Notification 10252611) Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
SRO 78	14052	40	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the normal valve alignment and physical response for the primary containment when venting.	Consider adding a description of this containment response (as well as response if nitrogen is isolated). (Notification 10254514) Cover this information during NRC post-exam review for current license class. (Notification 10254273)
SRO 79	32	0	"No operator action" was included in the stem twice. The test item required the candidate predict the containment response to an MSIV closure from full power with no operator action.	Remove the second "no operator action" from the question stem. (Notification 10254515) Cover this information during NRC post-exam review for current license class. (Notification 10254273)
SRO 80	36	20	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to know the purpose of the metal part of the Reactor Building (blowout panel protection for tornado loads). Description of these is not contained in the Containment student text (COR002-03-02).	Add description and function of the Reactor Building blowout panels to the Containment student text (COR002-03-02). Include discussion of the steam tunnel blowout plug as well. (Notification 10254516) Cover this information during NRC post-exam review for current license class. (Notification 10254273)

Question Number	Record Number	Pass %	Comment	Recommendation
SRO 84	48	40	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to evaluate provided Technical Specifications to determine testing requirements for a control rod drive mechanism.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
SRO 86	42	20	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to recall from memory the bases (plant design) for the jet pumps. Existing objectives require this knowledge from memory.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)
SRO 91	47	40	There is nothing technically or psychometrically wrong with this test item. The test item required the candidate to recall from memory whether to use SGT during initial containment purge and why/why not.	Cover this information during NRC post-exam review for current license class. (Notification 10254273)

2003 CNS NRC **Reactor Operator** Examination

Question Number	Record Number	Description	Source	Cognitive Level	References Required
1	1111	COR0010102, AC Electrical Distribution	Direct	3	
2	14036	COR0010102, AC Electrical Distribution	New	2	
3	1099	COR0010102, AC Electrical Distribution	direct	2	
4	14035	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES	New	2	
5	14670	COR0011402, OPS MAIN TURBINE	direct	1	
6	19124	COR0011802, OPS Radiation Monitoring	direct	2	
7	5084	COR0012002, OPS Reactor Water Cleanup	modified	1	
8	14043	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM	New	1	
9	14045	COR0013001, HWC Gas Generation System	New	2	
10	2931	COR0020302, CONTAINMENT	direct	2	
11	10081	COR0020302, CONTAINMENT	direct	3	
12	5155	COR0020402, CONTROL ROD DRIVE HYDRAULICS	direct	2	
13	14040	COR0020602, CORE SPRAY	New	2	
14	14050	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	New	1	
15	19090	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	3	
16	1507	COR0020802, Diesel Generators	direct	1	
17	14032	COR0021102, OPS High Pressure Coolant Injection (HPCI)	New	2	
18	19051	COR0021102, OPS High Pressure Coolant Injection (HPCI)	direct	2	
19	16513	COR0021402, Digital Electro-Hydraulic Control	direct	2	
20	1058	COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	2	
21	5425	COR0021602, OPS NUCLEAR PRESSURE RELIEF	New	3	
22	5608	COR0021602, OPS NUCLEAR PRESSURE RELIEF	direct	2	
23	14679	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM	direct	2	
24	14451	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)	direct	1	
25	14027	ACD0070307, Thermal Hydraulics (GP)	New	3	Calculator, Steam Tables, GFES Formula sheet.
26	2521	COR0021902, REACTOR EQUIPMENT COOLING	direct	2	
27	14039	COR0021902, REACTOR EQUIPMENT COOLING	Modified	2	
28	14024	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
29	1208	COR0022102, REACTOR PROTECTION SYSTEM	Modified	1	
30	19096	COR0022202, REACTOR RECIRCULATION	direct	1	
31	1744	COR0022302, RESIDUAL HEAT REMOVAL	direct	1	
32	4029	COR0022302, RESIDUAL HEAT REMOVAL	modified	2	
33	14028	COR0022302, RESIDUAL HEAT REMOVAL	New	1	
34	2127	COR0022602, OPS ROD WORTH MINIMIZER	direct	1	
35	14033	COR0022802, OPS STANDBY GAS TREATMENT	New	2	
36	14025	COR0022902, STANDBY LIQUID CONTROL	New	1	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
37	5348	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM	direct	1	
38	19084	COR0023102, TRAVERSING IN-CORE PROBE	direct	1	
39	14004	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL	New	1	
40	14014	COR0023402, Alternate Shutdown (LO)	direct	1	
41	8970	INT0070501, OPS Introduction to Technical Specifications	modified	3	
42	3995	INT0070502, CNS Tech. Spec. 3.1, Reactivity Control Systems	direct	2	
43	14048	INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	1	
44	14042	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS	New	2	T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.
45	6226	INT0320128, CNS Abnormal Procedures (RO) Containment	direct	1	
46	5247	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
47	13407	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
48	14000	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	New	2	
49	2403	COR0020902, Digital Electro-Hydraulic Control	direct	1	
50	19034	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	EOP Graphs
51	14001	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	New	2	
52	16472	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	direct	2	
53	16485	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM	Modified	3	EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs with entry conditions and cautions removed.and EOP Graphs.
54	54	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM	New	1	
55	14047	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.
56	5268	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	modified	2	EOP-3A with entry conditions and cautions removed.

Question Number	Record Number	Description	Source	Cognitive Level	References Required
57	5332	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	2	EOP-3A with cautions and entry conditions removed.
58	5333	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	1	
59	14023	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	1	EOP-3A with cautions and entry conditions removed.
60	19082	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-3A with cautions and entry conditions removed, 5.9H2O2.
61	5730	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	modified	1	
62	14019	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	EOP 5A with entry conditions and cautions removed.
63	14021	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	
64	16483	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	2	
65	19068	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	1	
66	14046	INT0080618, EOP AND SAG GRAPHS	New	2	EOP Graphs
67	4243	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)	direct	1	
68	12215	INT0320102, CNS Administrative Procedures Site Services Proc	direct	2	
69	12209	INT0320103, CNS Administrative Procedures Conduct of Operati	direct	2	
70	13673	INT0320104, CNS Administrative Procedures General Operating	direct	1	
71	16796	INT0320126, CNS Abnormal Procedures (RO) Cooling Water	direct	2	
72	14426	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum	direct	2	
73	4214	INT0320134, OPS CNS Abnormal Procedures (RO) - Fire	direct	1	
74	5127	SKL0080101, WATCHSTANDING PRINCIPLES	direct	1	
75	2167	RO only COR0010502, FIRE PROTECTION SYSTEM	direct	1	
76	1290	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYS	direct	1	
77	5092	RO only COR0010602, FUEL POOL COOLING AND DEMINERALIZING SYS	direct	1	
78	3724	RO only COR0010802, HEATING, VENTILATION, AIR CONDITIONING	direct	3	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
79	14041	RO only COR0011602, Off Gas	New	1	
80	3973	RO only COR0011602, Off Gas	direct	1	
81	3085	RO only COR0020102, AVERAGE POWER RANGE MONITOR	direct	1	
82	18311	RO only COR0020202, OPS CONDENSATE AND FEED	direct	2	
83	14053	RO only COR0020202, OPS CONDENSATE AND FEED	New	2	
84	3302	RO only COR0020302, CONTAINMENT	Modified	2	
85	14038	RO only COR0020402, CONTROL ROD DRIVE HYDRAULICS	New	1	
86	14496	RO only COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	2	
87	14034	RO only COR0020802, DIESEL GENERATORS	New	1	
88	3286	RO only COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	1	
89	1238	RO only COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
90	2816	RO only COR0022302, RESIDUAL HEAT REMOVAL	Modified	3	
91	19100	RO only SKL0124223, OPS RESIDUAL HEAT REMOVAL	direct	1	
92	5401	RO only COR0022402, ROD BLOCK MONITOR	direct	1	
93	5070	RO only COR0022902, STANDBY LIQUID CONTROL	direct	1	
94	16441	RO only INT0070501, OPS Introduction to Technical Specificat	direct	2	
95	14003	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	3	T.S. 3.0 section and bases, T.S. LCO 3.3.1.1 and bases.
96	14026	RO only INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	2	T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases.
97	19123	RO only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT	direct	1	
98	16466	RO only INT0320104, CNS Administrative Procedures General Op	direct	3	
99	13406	RO only INT0320124, CNS Abnormal Procedure (RO) Reactor Reci	direct	2	
100	12222	RO only INT0320115, CHEMISTRY PROCEDURES	Direct	2	
	58	**Direct**		**45**	**Memory (Cognitive 1)**
	10	**Modified**		**44**	**Comprehension (Cognitive 2)**
	32	**New**		**11**	**Analysis (Cognitive 3)**
	100	**total**			

2003 CNS NRC **SENIOR** Reactor Operator Examination

Question Number	Record Number	Description	Source	Cognitive Level	References Required
1	1111	COR0010102, AC Electrical Distribution	Direct	3	
2	14036	COR0010102, AC Electrical Distribution	New	2	
3	1099	COR0010102, AC Electrical Distribution	direct	2	
4	14035	COR0011302, OPS MAIN GENERATOR AND AUXILIARIES	New	2	
5	14670	COR0011402, OPS MAIN TURBINE	direct	1	
6	19124	COR0011802, OPS Radiation Monitoring	direct	2	
7	5084	COR0012002, OPS Reactor Water Cleanup	modified	1	
8	14043	COR0012402, OPS TURBINE EQUIPMENT COOLING SYSTEM	New	1	
9	14045	COR0013001, HWC Gas Generation System	New	2	
10	2931	COR0020302, CONTAINMENT	direct	2	
11	10081	COR0020302, CONTAINMENT	direct	3	
12	5155	COR0020402, CONTROL ROD DRIVE HYDRAULICS	direct	2	
13	14040	COR0020602, CORE SPRAY	New	2	
14	14050	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	New	1	
15	19090	COR0020702, OPS DC ELECTRICAL DISTRIBUTION	direct	3	
16	1507	COR0020802, Diesel Generators	direct	1	
17	14032	COR0021102, OPS High Pressure Coolant Injection (HPCI)	New	2	
18	19051	COR0021102, OPS High Pressure Coolant Injection (HPCI)	direct	2	
19	16513	COR0021402, Digital Electro-Hydraulic Control	direct	2	
20	1058	COR0021502, NUCLEAR BOILER INSTRUMENTATION	direct	2	
21	5425	COR0021602, OPS NUCLEAR PRESSURE RELIEF	New	3	
22	5608	COR0021602, OPS NUCLEAR PRESSURE RELIEF	direct	2	
23	14679	COR0021702, PLANT MANAGEMENT INFORMATION SYSTEM	direct	2	
24	14451	COR0021802, OPS Reactor Core Isolation Cooling (RCIC)	direct	1	
25	14027	ACD0070307, Thermal Hydraulics (GP)	New	3	Calculator, Steam Tables, GFES Formula sheet.
26	2521	COR0021902, REACTOR EQUIPMENT COOLING	direct	2	
27	14039	COR0021902, REACTOR EQUIPMENT COOLING	Modified	2	
28	14024	COR0022002, OPS REACTOR MANUAL CONTROL SYSTEM	New	1	
29	1208	COR0022102, REACTOR PROTECTION SYSTEM	Modified	1	
30	19096	COR0022202, REACTOR RECIRCULATION	direct	1	
31	1744	COR0022302, RESIDUAL HEAT REMOVAL	direct	1	
32	4029	COR0022302, RESIDUAL HEAT REMOVAL	modified	2	
33	14028	COR0022302, RESIDUAL HEAT REMOVAL	New	1	
34	2127	COR0022602, OPS ROD WORTH MINIMIZER	direct	1	
35	14033	COR0022802, OPS STANDBY GAS TREATMENT	New	2	
36	14025	COR0022902, STANDBY LIQUID CONTROL	New	1	
37	5348	COR0023002, SOURCE RANGE MONITOR SUBSYSTEM	direct	1	

Question Number	Record Number	Description	Source	Cognitive Level	References Required
38	19084	COR0023102, TRAVERSING IN-CORE PROBE	direct	1	
39	14004	COR0023202, OPS REACTOR VESSEL LEVEL CONTROL	New	1	
40	14014	COR0023402, Alternate Shutdown (LO)	direct	1	
41	8970	INT0070501, OPS Introduction to Technical Specifications	modified	3	
42	3995	INT0070502, CNS Tech. Spec. 3.1, Reactivity Control Systems	direct	2	
43	14048	INT0070504, CNS Tech. Spec. 3.3, Instrumentation	New	1	
44	14042	INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant System (RCS	New	2	T.S. 3.0 section and bases, T.S. LCO 3.4.4 and bases, T.S. LCO 3.4.5 and bases.
45	6226	INT0320128, CNS Abnormal Procedures (RO) Containment	direct	1	
46	5247	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
47	13407	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	
48	14000	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	New	2	
49	2403	COR0020902, Digital Electro-Hydraulic Control	direct	1	
50	19034	INT0080605, OPS FLOWCHART 1A - RPV CONTROL/RPV PRESSURE	direct	2	EOP Graphs
51	14001	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	New	2	
52	16472	INT0080606, FLOWCHART 6A - RPV PRESSURE/POWER FAILURE-TO-SCR	direct	2	
53	16485	INT0080612, FLOWCHART 7B - RPV FLOODING FAILURE-TO-SCRAM	Modified	3	EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A EOP-1A, EOP-2A, EOP-2B, EOP-6A, EOP-7A and EOP Graphs
54	54	INT0080610, OPS FLOWCHART 7A - RPV LEVEL FAILURE-TO-SCRAM	New	1	
55	14047	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-1A and EOP-3A with entry conditions and cautions removed, EOP graphs.
56	5268	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	modified	2	EOP-3A with entry conditions and cautions removed.
57	5332	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	2	EOP-3A with cautions and entry conditions removed.
58	5333	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	direct	1	

2003 CNS NRC **SENIOR** Reactor Operator Examination

Question Number	Record Number	Description	Source	Cognitive Level	References Required
59	14023	INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	1	EOP-3A with cautions and entry conditions removed.
60	19082	INT0080613, FLOWCHART 3A - PRIMARY CONTAINMENT CONTROL	New	2	EOP-3A with cautions and entry conditions removed, 5.9H2O2.
61	5730	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	modified	1	
62	14019	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	EOP 5A with entry conditions and cautions removed.
63	14021	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	New	1	
64	16483	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	2	
65	19068	INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINMENT AND RAD	direct	1	
66	14046	INT0080618, EOP AND SAG GRAPHS	New	2	EOP Graphs
67	4243	INT0320101 CNS ADMINISTRATIVE PROCEDURES (RO)	direct	1	
68	12215	INT0320102, CNS Administrative Procedures Site Services Proc	direct	2	
69	12209	INT0320103, CNS Administrative Procedures Conduct of Operati	direct	2	
70	13673	INT0320104, CNS Administrative Procedures General Operating	direct	1	
71	16796	INT0320126, CNS Abnormal Procedures (RO) Cooling Water	direct	2	
72	14426	INT0320132, CNS Abnormal Procedures (RO) Off Gas/Vaccuum	direct	2	
73	4214	INT0320134, OPS CNS Abnormal Procedures (RO) - Fire	direct	1	
74	5127	SKL0080101, WATCHSTANDING PRINCIPLES	direct	1	
75	768	SRO Only COR0012102, Refueling	Modified	1	
76	5101	SRO Only COR0012102, Refueling	Direct	1	
77	5468	SRO Only COR0020902, Digital Electro-Hydraulic Control	New	1	
78	14052	SRO Only COR0020302, CONTAINMENT	New	3	
79	32	SRO Only COR0020302, CONTAINMENT	New	1	
80	36	SRO Only COR0020302, CONTAINMENT	New	1	
81	44	SRO Only SKL0124223, OPS RESIDUAL HEAT REMOVAL	New	1	
82	4002	SRO Only INT0070501, OPS Introduction to Technical Specific	Direct	2	
83	38	SRO Only INT0070501, OPS Introduction to Technical Specific	New	2	T.S. 3.0 section and bases, T.S. LCO 3.3.3.1 and bases, T.S. LCO 3.8.1 and bases, LCO 3.8.3 and bases.
84	48	SRO Only INT0070502, CNS Tech. Spec. 3.1, Reactivity Contro	New	2	T.S. 3.0 section and bases, T.S. LCO 3.1.3 and bases, 3.1.4 and bases.

Question Number	Record Number	Description	Source	Cognitive Level	References Required
85	114	SRO Only INT0070502, CNS Tech. Spec. 3.1, Reactivity Contro	Direct	1	T.S. 3.0 section and bases, 3.3.2.1 and bases.
86	42	SRO Only INT0070505, CNS Tech. Spec. 3.4, Reactor Coolant S	New	2	
87	16415	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Syste	Direct	1	T.S. 3.0 section and bases, T.S. LCO 3.6.1.3 and bases, 3.6.2.3 and bases, 6.1RHR.201.
88	24	SRO Only INT0070507, CNS Tech. Spec. 3.6, Containment Syste	New	1	
89	40	SRO Only INT0070509, CNS Tech. Spec. 3.8, Electrical Power	New	1	
90	46	SRO Only INT0070604, TRM - Fire Protection	New	3	TRM section 3.11 and 0.23 "CNS FIRE PROTECTION PLAN".
91	47	SRO Only INT0070702, ODAM Specifications	New	1	
92	30	SRO Only INT0080613, OPS FLOWCHART 3A - PRIMARY CONTAINMENT	New	1	EOP-3A with cautions and entry conditions removed.
93	34	SRO Only INT0080617, OPS FLOWCHART 5A - SECONDARY CONTAINME	New	1	EOP-1A and EOP-5A with entry conditions and cautions removed.
94	16569	SRO Only INT0080618, EOP AND SAG GRAPHS	Direct	2	All EOP graphs.
95	28	SRO Only INT0320123, CNS Abnormal Procedures (RO) Reactivit	New	3	
96	5668	SRO Only INT0320134, OPS CNS Abnormal Procedures (RO) - Fir	Direct	2	
97	9015	SRO Only INT0320136, CNS Abnormal Procedures (RO) Miscellan	Direct	1	
98	5760	SRO Only INT0320101, CNS Administrative Procedures Volume 0	Direct	3	
99	12218	SRO Only INT0320116, Health Physics Procedures	Direct	2	
100	9684	SRO Only SKL0110101, SRO Upgrade Self-Study Program	Direct	2	0.50, Outage Management Program.
	51	**Direct**		**45**	**Memory (Cognitive 1)**
	9	**Modified**		**44**	**Comprehension (Cognitive 2)**
	40	**New**		**11**	**Analysis (Cognitive 3)**
	100	**Total**			